A N N U A L · R E P O R T · 92

Safety Technology Institute

JOINT RESEARCH CENTRE

COMMISSION OF THE EUROPEAN COMMUNITIES

EUR 15055 EN

ANNUAL · REPORT · 92

Safety Technology Institute

JOINT RESEARCH CENTRE

COMMISSION OF THE EUROPEAN COMMUNITIES

EUR 15055 EN

Published by the
COMMISSION OF THE EUROPEAN COMMUNITIES

Directorate-General XIII
Information Technologies and Industries, and Telecommunications

L-2920 Luxembourg

LEGAL NOTICE

Neither the Commission of the European Communities nor any person acting on behalf of the Commission is responsible for the use which might be made of the following information.

Cataloguing data can be found at the end of this publication.

Luxembourg: Office for Official Publications of the European Communities, 1993

Printed in Italy

CONTENTS

EXECUTIVE SUMMARY

NUCLEAR ACTIVITIES

Reactor Safety

The year 1992 was the first of a new multiannual programme (1992-1994) and saw a further reduction in the number of reactor safety research activities of our programme. At present the work performed at the Safety Technology Institute (STI) is almost completely devoted to LWR severe accidents and in particular to the phenomena which could characterise a number of accident scenarios and the possible release of radioactivity to the environment. In spite of the limited size of the programme, a number of very important activities both experimental and theoretical are being carried out with considerable success.

In the area of Source Term the STI participation in the international Phebus-FP programme is very active and qualified. Phebus-FP, under way at Cadarache (F), is approaching a crucial phase because the first test FPTO with fresh fuel will be executed in 1993, after more than 5 years of preparatory work, and immediately after the second part (Block B) of the CEC-CEA agreement (Convention) for the execution of 5 in-pile tests with irradiated fuel will be started. In 1992 the STI contribution in this programme concerned the follow up of the project activities by the STI team detached to Cadarache, the participation in test precalculations and the coordination of partners contributions as well as the work for updating the test matrix, which will be approved by the Steering Committee in 1993.

An other important aspect of the Source Term research is the ongoing development of the informatic system ESTER which has been designed as an advanced mechanistic code package for Source Term evaluation and will be available to the Member States for Phebus-FP and severe accident sequence calculations. ESTER will include in a modern informatic structure a number of modules already developed and validated at national level or still under development. A number of Shared Cost Action (SCA) contracts, which are now being completed, have given essential support to this project and made the cooperation with the European Partners easier. The first pilot version of ESTER was released, after a workshop organised at Ispra in June 1992, to the laboratories involved in the Phebus-FP programme. Further developments, validation and maintenance of ESTER will be assured in the future by the STI team. As a part of the work on codes the STI team was also involved in a number of international benchmark calculations, performed in the frame of Phebus-FP or of CSNI groups (ISPs).

Still in the field of Source Term the STI has decided at the end of 1991 to start the design and construction of a large scale experimental facility called STORM for the investigation of aerosol deposition and resuspension in primary system components. This project is jointly sponsored by the JRC and ENEL (I) and is open to the cooperation of other Institutions. The construction will be terminated by the end of 1993.

The FARO programme was successfully going on even if difficult new technological problems have continuously to be solved. The number of Organisations interested in this programme was increased and probably new agreements will be signed in 1993. The year 1992 saw the execution of two tests, the first in April for the Jet Characterisation (150 kg of melt mass, no water), the second in July to obtain additional data on melt quenching into water, at constant volume, with intermediate melt mass (about 50 kg) and increased volume of the test section. From both tests very interesting results have been obtained even though not all the initial objectives were reached.

A considerable effort has been made to complete a detailed documentation of the first melt quenching test and to start the documentation of the second. Precalculations and post-test analysis have been performed using TEXAS II and RELAP5, while the 2D IFCI code (USNRC) is being implemented. Some of the European Partners participated in the test analysis with their codes.

The small-scale KROTOS programme for FCI investigation, was also successfully continuing. The tests with Aluminium Oxide with and without trigger have been completed in 1992. New tests with Uranium Oxide to which USNRC will contribute were prepared for 1993.

In the area of LWR thermalhydraulics the work on LOBI test documentation and analysis as well as the validation/improvement of the system code CATHARE-2 (collaboration contract with CEA) has been continued. A Seminar on the LOBI programme was successfully organized in April 1992.

Safeguards and Waste

In the frame of Fission Energy Safety substantial progress has been made in the Nuclear Island in setting up and finalizing the test facilities. The Performance and Training Laboratory (PERLA) has successfully undertaken all nuclear tests, required by the Licensing Body and is now regularly in operation. It has taken over all activities previously performed in Pre-PERLA in the field of Non-Destructive Assaying of Nuclear Material for safeguards purposes in the frame of both the JRC Specific Research Programme and in particular the Commission's Support activities for the Euratom Safeguards Directorate (ESD, DG XVII Luxembourg) and IAEA (DG I) Inspectorates.

The TAME Laboratory, entirely devoted to Inspectors' training on industrial scale tank volume and weight measurements in reprocessing facilities, has been assembled and is now ready for commissioning. At the time being a 12 m^3 input and a 0.25 m^3 output accountancy tank (for Pu nitrate) are available. For small scale training exercises, however in realistic "hostile" conditions, i.e. in presence of full α,β,γ activity levels, the hot cell facility PETRA is now ready to take up operation.

During 1992 indeed all tests with sealed sources and non-irradiated uranium have been successfully completed. The introduction of irradiated fuel to start with studies of radwaste treatment and conditioning alternatives will depend, however, on the availability of additional funding from outside the Frame Work Programme budget.

A particular item of the R & D activities in the safeguards area, besides ongoing instruments upgrading and improvement, has been an international workshop on calorimetry, held during 1992 in PERLA. The validity of this method as a complementary method for Pu accountancy has been emphasized. Another area of growing importance has turned out to be the combination and integration of different measurement techniques like High Resolution Gamma Spectrometry (HRGS) and Passive Neutron Counting (PNC), by appropriate elaboration and interpretation software. Beyond the resulting performance in quantitative measurement of NDA instruments, the fissile material verification planning and the statistical data processing of verification results are to be placed into an integrated safeguards methodology and information treatment system, a problem on which STI in working now.

A new field in which the safeguards team of STI has been involved with collegues from ISEI is the integration of NDA methods with Containment/ Surveillance (C/S) systems to develop eventually unattended Safeguards systems which allow a round the clock inspection without unacceptable burdens for inspectorates manpower.

In the frame of the Commission's Support activities STI has given assistance to the IAEA and in particular the ESD Inspectorates as defined in specific task sheets for a series of NDA instruments either already in use in field by the inspectors or still in a phase of development and qualification mainly in PERLA. This facility with the availability of industrial like fuel material samples (PERLA standards) played a central role in the continuously growing requests for training courses, Physical Inventory Verification (PIV) and calibration exercices. A total of 15 such courses of 1 to 2 weeks duration have been organized by the safeguards staff in the Nuclear Island.

The increasing need of non-destructive monitoring of radwaste packages (by operators for classification and various authorities for control) has led to a reinforced effort at the Nuclear Island in this field. The Time Correlation Analyser (TCA) for passive neutron monitoring of Pu waste has been further upgraded and will be put on the market in the frame of a commercial agreement with a private firm. On request of the Safeguards Inspectorate a measurement campaign of waste drums in a MOX fuel fabrication plant has been performed with the prototype instrument with very satisfactory results.

For the measurement of uranium in fuel fabrication plants the active neutron interrogation system Phonid 3bis has been tested at a UK facility demonstrating its applicability. Consequently, theoretical and experimental verifications have been agreed and performed with interested operators for developping an industrial scale U-monitor.

For the classification of waste packages (220 dm^3

drums), according to the technical guidelines issued by the regulatory bodies, the quantitative assessment of gamma activity is required. For this purpose a system was set up scanning the activity of gamma emitters and the matrix density in a series of 'subsequent' sectors of the package. The software for data elaboration on PC was completed. Industrial operators are going to perform test measurements on this system.

Fusion

The European Tritium Handling Experimental Laboratory (ETHEL) has been taken over by JRC from the Main Contractor after conclusion of the functional tests of the various systems. Under JRC responsability, the testing of systems for commissioning purposes has been initiated. Assembling of experimental loops in the glove-boxes has proceeded in parallel. In close co-operation with the German Tritium Laboratory at Karlsruhe (TLK), a joint tritium control methodology and related procedures have been developed and submitted to the ESD. Common tests with TLK were undertaken with highly satisfactory results on the JRC calorimeter and gas chromatographic system, the latter entirely developed in STI. The excellent performance of regenerable activated iron (industrial product) for water decomposition has raised considerable interest in such a process as an alternative to electrolysis for highly tritiated water.

The improvements in qualitative and quantitative separation performance obtained through the preparation and characterisation of new substrates of the zeolite type and other inorganic materials have fully justified the ongoing research efforts in this field, which coincidently are of high interest for application to all kinds of gas-solid separation processes.

NON-NUCLEAR ACTIVITIES

Industrial Hazards

The focus of this activity is placed on the assessment, improvement and harmonization of safety methodologies for chemical plants. The FIRES facility for studying the safety of batch chemical reactors has become fully operational, and typical incidents regarding the nitration of toluene have been investigated. The venting code RELIEF has reached an advanced stage of development. The MPMC venting facility continues to produce results of industrial relevance and is now being directed towards large scale studies. Experimental data and design methodologies have been contributed to a number of international working groups (ISO/ Technical Committee 185, DIERS, and DECHEMA). In the field of dense gas dispersion the three-dimensional computer code ADREA-HF is under validation, and a one-dimensional shallow layer model is near completion. A computer code for reacting gas flows is under development, which will allow to simulate explosions under confined and unconfined conditions.

The JRC ERCOFTAC Pilot Centre has formulated a proposal for a collaboration network of 9 EC member states on the "Harmonization in Atmospheric Dispersion Modelling Systems for Regulatory Purposes", with particular reference to models related to industrial sources. European collaboration networks on runaway reactions and venting are being organized.

Reference Methods for the Evaluation of Structural Reliability

The activities in the field of structural safety are largely centered around the use of the new reaction-wall facility, now named ELSA, which entered into operation in May 1992 and was officially inaugurated on October 16 in the presence of

Authorities and technical delegations from the Member States of the Community.

The facility will initially be used for prenormative research in support of Eurocode n° 8 (EC8), the European design code for structures in seismic areas. It will also be available to industry for testing innovative design concepts and prototypes.

A first integrated research programme, which concerns the earthquake behaviour of civil engineering structures, has been set up in close collaboration with Directorate General III of the European Commission, the EC8 expert group, and a number of research organisations in the Member States grouped into the European Association of Structural Mechanics Laboratories.

The execution of this integrated programme will be performed jointly by the Association and the ELSA team now grouped within a scientific network under the Human Capital and Mobility (HCM) programme. Access to ELSA will also be facilitated through its recent recognition as a "Large Installation" under the HCM programme.

During the year 1992, the experimental activity in the ELSA laboratory consisted mainly of testing the proper implementation of the pseudo-dynamic (PSD) test method by performing first large-scale experiments on a reinforced concrete beam and a 3-storey steel frame. Very encouraging results were obtained from these calibration experiments which indicated the ability of the PSD method as implemented at ELSA to accurately reproduce the actual dynamic response of the tested structures. In parallel with the above validation experiments, design work has been performed to prepare a PSD test to be executed in 1993 on a 4-storey reinforced concrete frame.

As a support to the research in structural dynamics, tests at various strain rates were performed to characterize the dynamic behaviour of concrete and steel. A number of Hopkinson's bar tests under mono- and bi-axial loading were also performed on ceramic and composite materials in the framework of collaboration contracts with industry.

In the area of computational mechanics, activities have concentrated on the further development of local and global models for predicting the nonlinear cyclic behaviour of steel, concrete and masonry structures. A particular emphasis was placed on semi-global and global member models with the view of allowing simulation of the nonlinear dynamic response of complete frame structures. The resulting numerical models and tools are being incorporated in the CASTEM-2000 finite element system developed by CEA-Saclay and now used as the basic computer programme system for our earthquake engineering projects.

Further refinements have also been made in the PLEXIS-3C code for fast dynamic fluid-structure problems developed in collaboration with CEA-Saclay. In particular, a new version of the code has been produced with improved modelling features and a post-processing interface to CASTEM 2000. Other PLEXIS-3C activities were performed in the framework of a third-party work contract.

1 SPECIFIC PROGRAMMES

1.1 REACTOR SAFETY

The Reactor Safety Programme at the STI is focussed on the study of phenomena which characterise accidents especially to so-called Severe Accidents in Light Water Reactors (LWR). In 1992 the programme was subdivided into the following main chapters:

- *Source Term, where the main effort is concentrated on the participation in the Phebus-FP programme, including the development of the European source term code ESTER and the experimental project on the investigation of aerosol resuspension phenomena STORM,*
- *FARO an international programme on the interaction of molten fuel with coolant and reactor structures,*
- *Thermal hydraulics which includes activities related to the LOBI test analysis.*

The relevant results are described below.

1.1.1 SOURCE TERM

Introduction

The STI research in this area includes several activities which are performed in collaboration with many partners from EC and non-EC countries. The work in this field has the objective of contributing to a more realistic evaluation of the source term based on a large consensus and international cooperation.

The participation in the international Phebus-FP programme is the most important effort in this area, due to the resources involved, the ambition of the overall programme and the number of partners. In addition to the financial participation, defined by the CEA-CEC "Convention" signed in 1988, the STI staff is directly involved in the project work at Cadarache (particularly in advanced instrumentation and material problems), in the test definition and analysis and in the coordination of the partners work.

The activity was part of Commission's Shared-Cost-Action (SCA) programme on Source Term which allowed important contributions from the Member States in support of the Phebus-FP programme and on complementary research aspects. A number of these contract are still under way or have been terminated in 1992. In particular the SCAs gave substantial support to the development of the ESTER code package, based on the best available source term calculation models and informatic techniques, to the development of the VICTORIA code and other modules which will be part of the ESTER system, to experimental and theoretical studies on fission product and aerosol chemistry (Falcon project) etc.

Unfortunately in the present programme 1992-1994, due to the reduction of resources, the SCA programme has not been extended and this is asking for an additional effort from the STI which has to take over a part of the work previously performed outside, in particular the development/validation of the ESTER system which on the long term will be one of the most relevant outcome of the STI programme.

Among the activities on aerosol physics related to the Phebus-FP programme, an important project, which foresees the construction of large-scale separate effect test facility called STORM, for the study of aerosol deposition/resuspension, has been started by the STI in collaboration with ENEL (I) in 1991. The resuspension phenomenon could have an important impact on source term and up to now data and models available are very unsatisfactory. The STORM project is planned to come into operation by the end of 1993.

Phebus-FP Programme

The Phebus Committee met twice in 1992, and took some important decisions regarding the tests. For the first test, FPT-0, now planned for execution in the summer of 1993, it was decided that in order to favour the mobilisation of iodine during the so-called chemistry phase of the containment portion of the experiment the sump water will be given an initial pH of 5, and that buffering substances will be added to maintain this acid pH constant throughout the experiment. The relatively low levels of radiation and low concentrations of iodine expected in the sump together with uncertainties about the detectability limit for atmospheric iodine species (particularly I_2, CH_3I) were important factors in this decision.

A second Committee decision concerned the second experiment, FPT-1, planned for mid-1994. According to the original test matrix there were to be two options for the circuit portion of the test geometry, a (hot) steam generator-tube as in FPT-0 and a so-called "minimum retention line", which proceeds from the head of the rising position of the circuit to the containment vessel in the most direct way possible within the constraints of the caisson design. According to the original plan one option or the other was to be chosen depending on the measured retention of fission products in test FPT-0. The need to order circuit components well in advance required a decision in 1992 however, and following the results of calculations which indicated that the fission product retention was not greatly different (factor of 1.5 to 3) with the two options a compromise configuration was decided upon, which is identical to that of FPT-0 up to the exit of the steam generator tube, but takes a simplified path with fewer bends from there to the containment vessel. Simultaneously, the instrumentation station "G" has been moved closer to the vessel and should therefore give a better indication in FPT-1 of the input from the circuit to the vessel.

Il line with the trend of 1991 towards a simplification of the text matrix a joint JRC/CEA team has been charged by the Committee with proposing a revised test matrix.

The SAWG held three meetings during 1992, in which the test protocol for FPT-0 was drawn up. This document describes the test geometry and the initial and boundary conditions. It also specifies the steps to be taken in the case of certain malfunctions (e.g. loss of control of the circuit heaters), to ensure a successful test, the strategy to be adopted for the control of the bundle neutronic power and steam flow and the temperature of the condenser in the containment vessel, and specifications for the operation of the various sequential sampling devices such as impactors and May-packs. On the basis of this document the experimental team will draw up the detailed operating instructions for the performance of the test.

The SAWG has also drawn up a provisional test protocol for test FPT-1 sufficiently detailed for exploratory and sensitivity calculations, and an outline protocol for test FPT-2. A series of Task Forces has been set up, with specific mandates and finite lifetimes:

- Paints (specification, characterisation, modelling)
- Thermal Resuspension (in or out of pile procedures, phenomenology)
- Complex Structures (effect of e.g. steam driers in accident sequences; scaling and study in Phebus)
- Boric Acid (presence and effects in reactor accidents, mode of introduction in Phebus)
- Late-phase degradation (phenomenology, feasibility of study in Phebus)
- Depressurisation and Sprays (reactor phenomenology, most attractive aspects for study in Phebus)

Project Progress

At the end of the civil engineering construction period the contractors left the reactor site between January and April, 1992. A number of studies were pursued to document the earthquake-resistant design of several smaller components. Testing and commissioning (T&C) of buildings and ventilation systems were concluded towards the end of the year.

All so-called conventional circuits have been installed and their T&C began in September. This work concerned:

- reactor primary circuit,
- reactor secondary circuit,
- pool surface warm water layer ("blanket") system,
- intermediate water loops,
- cooled water ("ice water") loops,
- demineralized water supply,
- high pressure experimental water ("LOCA") loop,
- various compressed air supplies,
- liquid waste system,
- organic liquid loops for vessel, sump and condenser temperature control,
- tap water supply.

T&C are scheduled to end in May 1993 with full power tests of the new reactor cooling circuits.

Installation of experimental circuits and their instrumentation was ongoing at the end of the year inside the FP caisson (***Figs. 1.1*** to ***1.4***). Their final testing is planned for April 1993. The basic layout options for the circuits of the second test, FPT-1, were specified.

Various shielded handling containers and facilities have been supplied during the year. Installation and T&C of the large test train examination facility "PEC" are scheduled for April 1993. The shielded containers for waste and for the containment vessel condensers will not be available for the first test. They can be replaced, however, by alternative temporary devices and methods, due to the lower remaining radioactive inventory of FPT-0.

The in-pile test train for FPT-0 was completely manufactured and assembled during the year, and is expected to be supplied to the site as required. The same holds for a number of auxiliary devices for hydraulic and neutronic measurements. Design studies for the FPT-1 test train have been prepared. They include options with an inner high density channel of ThO_2 or ZrO_2. The main test train insulation ("shroud") consists of 40% porosity Zirconia developed through an R&D programme concerning novel fabrication methods and high temperature thermophysical data, and taking recent safety requirements into account. This work is expected to continue through a major part of 1993 (***Fig. 1.5***).

The fuel rods for FPT-1, pre-irradiated in BR 3, to about 27 MWd/t, have been examined by TUI Karlsruhe, and the first 49 rods were shipped to Cadarache. All are technically sound, with axial/radial distributions of fission products and heavy nuclei as predicted. Fission gas inventory/pressure and fuel burn-up are within a few percent of calculated data. The tests at Karlsruhe on the remaining rods will continue in 1993.

The instrumentation plan for FPT-0 was finalized during the year, together with specifications for post-test analyses of the FP samplers and for post-irradiation examinations of the test fuel bundle. A draft instrumentation plan for FPT-1 was ready by the end of the year.

An important effort was invested into the licensing procedure for reactor and experiments. Six volumes of the formal Safety Report were transmitted to the Authorities during the first half of the year, followed

Fig. 1.1 Liquid sampling capsules (containment vessel)

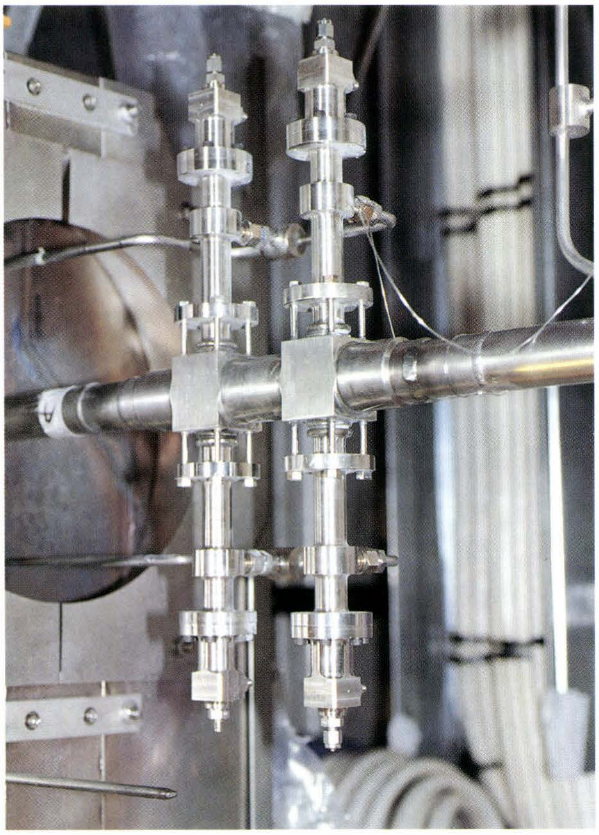

Fig. 1.2 On-line aerosol monitor OLAM

by a large number of additional data requested during the preliminary review process. Formal meetings of the Safety Committee are scheduled for January and April 1993.

Test preparation continued as improved pre-calculations for FPT-0 became available, including instrument precalculation. The test protocole was practically finished by the end of the year, enabling detailed operating procedures to be written. The experimental facility thermal-hydraulic tests prior to FPT-0 will include conditions anticipated in future experiments.

The JRC team delegated to Cadarache remained fully involved in the day-to-day project activities, with specific contributions in the following areas:

- supply and maintenance of a data handbook for each experiment,
- development of consecutive experimental instrumentation plans,
- R&D of novel instruments and of the test train shroud,
- post-test analyses of FP samplers and post-irradiation examination of the test fuel bundle,
- telemanipulation and decontamination,
- test fuel characterisation.

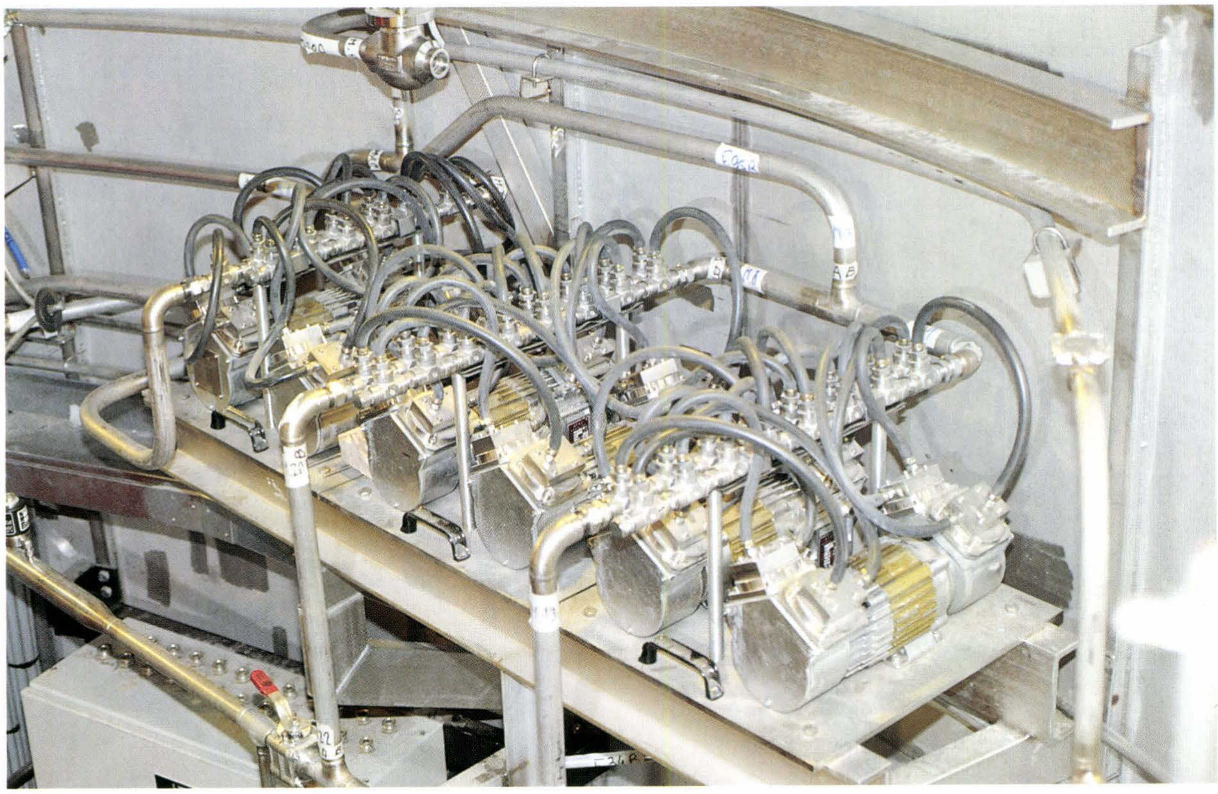

Fig. 1.3 Recirculation pumps between containment vessel and off-gas line

Fig. 1.4 Power manipulator inside the caisson

The overall planning for FPT-0 indicates that T&C and licensing are presently on the critical path towards the test. The schedule established end 1992 identified May-June 1993, as the most probable period for the first test.

Test Matrix Revision

In the meeting of March 1992 the Phebus-FP Steering Committee decided to undertake a revision of the Phebus FP Test Matrix in order to take account of the extensive preparatory work already performed and of new priorities suggested by the evolution of research on Source Term. In the following meeting of October 1992 the Steering Committee, after a discussion on a progress report, clarified better the objectives of such a revision, which are summarized in the following guide-lines:

- The overall approach in the definition of test objectives and conditions is "phenomena oriented" and attempts to refer to severe accident sequences which are limited to generating test conditions necessary to study these phenomena.
- The phenomena to be taken into consideration should reflect existing priorities in LWR Source Term evaluation.

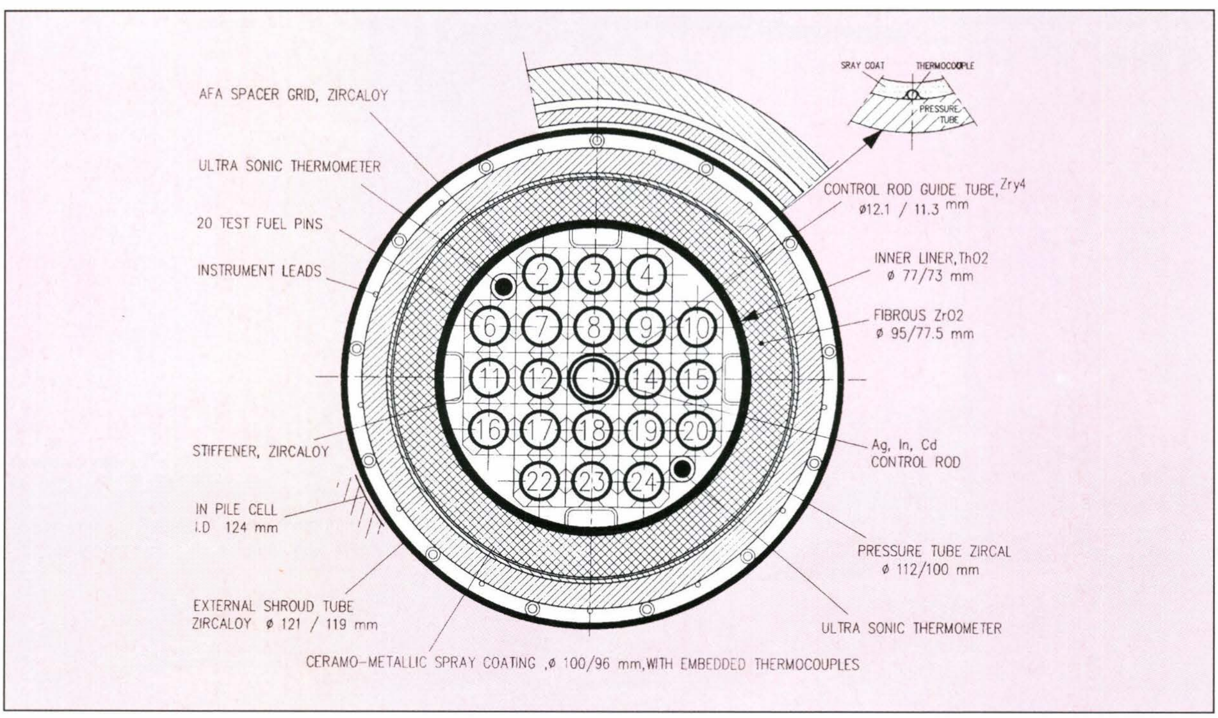

Fig. 1.5 FPT1-FPT2 in-pile test section: design study

- The proposed modifications should go in the direction of simplifying the tests and improving the feasibility.
- Phenomena which are more suitable to the studied out-of-pile should be withdrawn, taking also into account existing out-of-pile test programmes.
- Priority should be given to phenomena pertaining to existing PWRs, but attention should be paid also to Advanced Reactor concepts and BWRs.
- The Phebus FP results should be used primarily to validate computer codes.

The activity performed in the reporting period include the following actions:

- consultation of European organizations
- written inquiry of cosponsors
- assessment of results of Shared Cost Actions in support of Phebus FP
- discussions with the Phebus FP Project teams
- assessment of technical documentation produced in the frame of TG and SAWG activities.

In December 1992 a preliminary proposal of revision was released to the interested parties. At this stage of discussion, the main orientations can be summarized as follows:

- The components of the primary circuit concerning the V sequence, steam generator tube rupture and pool scrubbing under PWR conditions are deleted and the configuration of the primary circuit will be maintained simple in all the tests.
- The test matrix is divided in two parts, the first one of which includes three tests and covers the main objectives of the previous Test Matrix. These three tests are practically unchanged with respect to the former version.
- The fourth test is an open test with three possible options: 1) a back-up test in which phenomena not sufficiently clarified in the previous tests are repeated; 2) a high pressure test; 3) a test devoted to BWR and advanced fuel studies.
- The last two tests include new phenomena concerning late phases of PWR fuel degradation and corresponding release of less volatile FPs and transport under conditions of enhanced clad oxidation due to air cooling.

The proposed revision has brought some simplification in the primary circuit, but the new phenomena introduce very challenging problems of feasibility, safety and instrumentation as well as some modelling needs. These problems have to be assessed by the Project teams before the revised Test Matrix can be proposed to the Steering Committee in March 1993.

Neutronics Calculations

As already mentioned at previous occasions, neutronics calculations for Phebus require a detailed 3-D analysis because of the very heterogeneous structure of the reactor assembly containing the test loop. For this reason the JRC Ispra performed neutronics calculations using its Monte Carlo code KENEUR which allows for a very detailed 3-D modelling of the experimental arrangements and at the same time makes use of a unique feature to calculate (small) perturbation effects by a sophisticated low variance correlation technique. With this approach complementary to deterministic calculations carried out at Cadarache and elsewhere coupling factors for water and steam cooling, space dependent control rod worths, as well as the optimal position of the fission chambers for detecting Ag, In, Cd and fuel melt-down were determined in previous years. Now with reflector savings, a revision of control rod worth and axial fission power distributions are dealt with in a refined geometrical mock-up.

The reflector saving

In order to increase the Phebus reactivity reserves the effect of replacing the first part of the "infinite" water reflector (behind the core vessel) by graphite was studied. In these calculations the exact structure of such a reflector consisting of one or more layers of graphite blocks (5.0 cm thickness, slightly contaminated by a boron equivalent) canned in zirconium (0.1 cm) and separated by 0.3 cm water gaps was taken into account. It turns out that a reactivity gain of $\Delta\rho \cong 0.019$ can be achieved for 5 cm of graphite, $\Delta\rho \cong 0.023$ for 10 cm and $\Delta\rho \cong 0.030$ for 30 cm.

The control rod worth

Over the period of the experiments a relatively large burn-up is expected and therefore the control devices must be able to act over a wide reactivity range. The complete withdrawal of all control rods results in a $\Delta\rho \cong 0.19 \pm 0.04$ and a coupling factor change of $\Delta(C.F.) = 16$ (control rods out).

Compared with CEA results the control rod worth calculated by the JRC seems to be too large. Therefore a detailed study of geometrical self-shielding effects was carried out. To this end control rods made-up of the real model of hafnium pins were compared with homogenized assemblies. It turned out that there was practically no difference between the homogenized and the heterogeneous model. The disagreement is therefore most probably caused by the difference of the hafnium cross sections.

The Axial Fission-Power Distribution

The calculated axial fission-power distribution showed differences with gamma-scan measurements performed at Cadarache. This quantity is of great importance in thermo-hydraulic calculations. Therefore considerable efforts have been dedicated to elaborate a more refined 3-D model considering all the geometrical details at the top and bottom end of the test bundle which might influence the fission distribution. Unfortunately the transfer of the reactor code systems from the main-frame to workstations has been associated with unexpected technical difficulties and delays.

FPT-0 Test Preparation

As explained in the 1991 report, the objectives of the first Phebus test have been considerably simplified compared with the definition in the original test matrix. The steam flow to the rod bundle (20 fuel rods plus one control rod) is kept high enough to exclude starvation conditions, and there is consequently little predicted eutectic formation. Degradation conditions are severe, however, with a target of 20% of the fuel heated above the melting temperature. The resulting thermal loads to the shroud are challenging. A revised shroud design was made available by the Project in February 1992 with the appropriate material properties, in particular the thermal conductivity and thermal expansion coefficient (as functions of temperature). As described in the section below, an extensive series of calculations has been performed by CEA, the JRC and the international Partners (coordinated by the JRC) to evaluate the behaviour of the bundle plus shroud under the planned neutronic and steam injection transients [1,2,3]. One concern is controlling the bundle transient in the absence of the thermocouples within the bundle and in the inner layers of the shroud, since these instruments are expected to fail above about 1400°C. A control strategy has been developed based on internal and external temperature measurements at a series of relatively low temperature plateaux followed by reliance on the surviving shroud thermocouples as the temperature transient proceeds.

A further constraint upon the bundle transient is the outlet temperature from the bundle. The circuit section immediately above the bundle is unheated and must not be allowed to become too cold to avoid strong trapping of the fission product vapours emitted by the bundle. In practice this imposes a lower limit upon the steam flow. On the other hand the circuit segments above the unheated portion are heated by electrical circuits which must not be allowed to become too hot, to avoid burnout. This constraint translates into an upper limit on the steam flow.

A final constraint on the steam flow (a lower limit) is the objective of no steam starvation. One goal of the bundle calculation has been to find a combination of steam flow and neutronic power to meet all the objectives at once. Somewhat differing strategies result from the use of different codes, reflecting the existing uncertainty regarding oxidation rates, degradation processes etc., as well as uncertainties more specifically linked to Phebus such as those in the shroud properties.

Possibilities for variation in the circuit conditions are small, and circuit calculations have mostly been directed to evaluating the consequences for FP transport and deposition of the strategy adopted for the bundle, including benchmark studies concerning the expected release histories for fission products, control rod constituents and structural materials, and to resolving the differences between the retentions predicted with various codes [4].

Concerning the containment it was decided to split the scenario into an aerosols phase, during which fission products and other materials arrive from the circuit and are deposited by settling, diffusiophoresis and other mechanisms, a washing phase to transfer settled materials to the sump, and a chemistry phase during which the main interest is in the chemical transformation and mobilisation of iodine and other radiologically important elements under the combined influence of radiolysis, the paint coupons in the sump and atmosphere and the chemical contents of the sump water. A series of code comparison exercises for the aerosol phase stressed the need for specific thermal hydraulic tests to evaluate the effective heat and mass transfer coefficients between sump, wall and condenser. A separate series of preparatory calculations has been performed to plan these thermal-hydraulic tests, which will take place early in 1993 [5]. As regards the chemistry phase, a benchmark exercise involving several participants and codes is reported below which has helped to identify the principal uncertainties in iodine chemistry prediction and to guide the separate effects experiments being performed to characterise the Ripolin paint to be used in the containment of FPT-0 (coupon in the sump, and complete painting of the dry and wet parts of the condenser structure).

BUNDLE CALCULATIONS FOR FPT-0

The main objective of the bundle region in Phebus test FPT-0 is to study the release, retention and transport of fission products from a degraded core in an oxidizing environment. The degradation aimed for is rather ambitious: 20% of the uranium dioxide fuel should reach a temperature higher than its melting point of around 3050K and these temperatures should be held for half an hour. Boundary conditions were proposed to reach these objectives and calculations performed at JRC and elsewhere [3,6] using the most advanced European and American core degradation computer codes were performed to check whether these boundary conditions had been correctly specified. The codes agreed that the oxidation of zircaloy cladding would at no time consume all the steam and this was considered a sufficient condition for the atmosphere to be classed as oxidizing. They also agreed that, provided that the conductivity of the shroud surrounding the bundle did not deviate from its expected values, then at least some of the fuel will reach the melting temperature of uranium dioxide.

The excess of steam will lead to fast oxidation early in the transient. Probably the oxidation will be so fast that the clad will oxidize before it melts and so there will be little dissolution of uranium dioxide by molten zircaloy. The detailed degradation behaviour during later phases of the experiment is harder to predict because the codes have not been validated for such severe degradation. Of the previous bundle experiments in Europe and elsewhere. Phebus-SFD, and CORA only went up to about 2700K and PBF and LOFT were terminated as soon as 3000K was reached.

During the early part of 1992 the reference scenario was changed again. Firstly the experimental team wanted a series of temperature plateaux in the initial part of the transient to calibrate their instruments and secondly the design of the shroud had changed. A new set of power and flow conditions were then specified which produced suitable plateaux but left the subsequent heat-up unchanged.

A new feature of the shroud design was the presence of two annular gaps separating the layers of porous zirconia as shown in ***Figs. 1.6*** and ***1.7***. At Ispra calculations were performed with ICARE2 to see whether uncertainties in the width of these gaps would make the experiment harder to control or not.

Fig. 1.8 shows the temperature at the bundle midplane for five cases. Case 1 is the base case, in case 2 the gaps in the shroud are replaced by porous zirconia, in case 3 the gap widths are doubled, in case 4 the conductivity of the porous zirconia is doubled and in case 5 it is halved. This clearly shows that the gaps strongly influenced the shroud's thermal resistance during the plateaux but later in the transient their influence is less and the conductivity of the porous zirconia has more effect. This is because thermal radiation means that at high temperature the thermal resistance of the gaps is reduced and because thermal expansion tends to close the gaps anyway. These calculations, described in [7], warned the team who will control

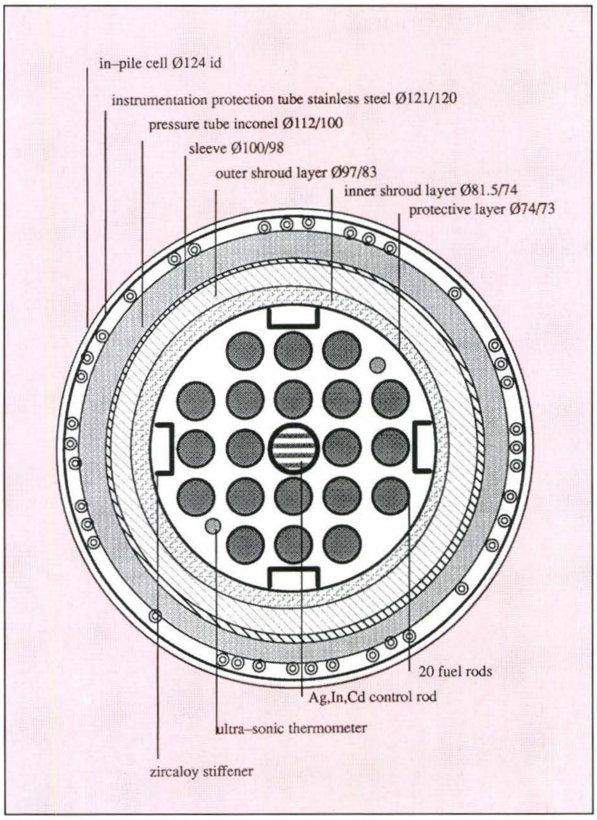

Fig. 1.6 New shroud design

the experiment that they should not rely on temperature measurements during the plateaux to predict the subsequent behaviour of the bundle.

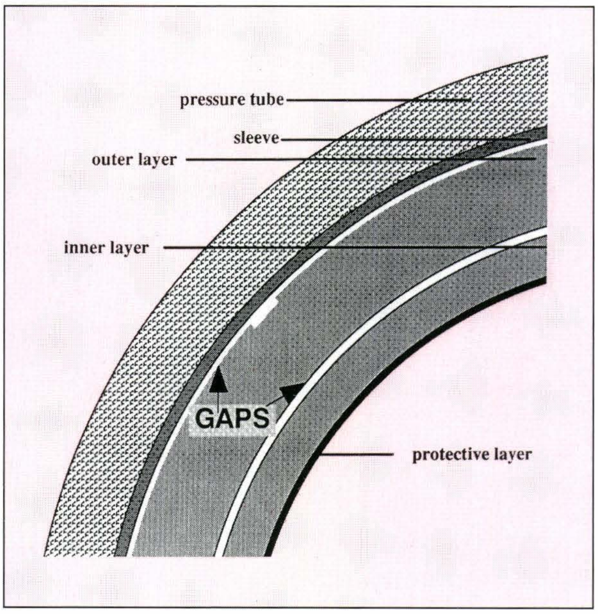

Fig. 1.7 New shroud design: gap

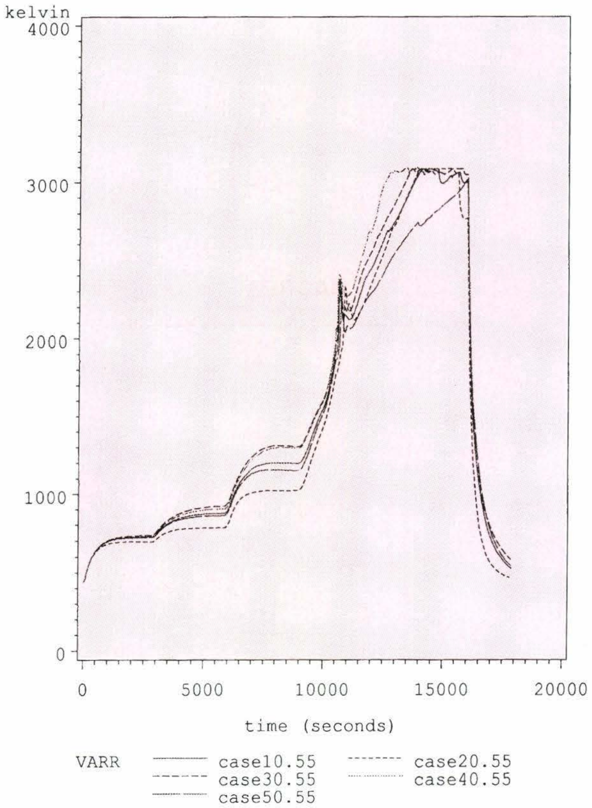

Fig. 1.8 Fuel temperature at 0.55 m elevation

JRC Ispra continued to coordinate and analyze calculations by other organizations involved with Phebus *Fig. 1.9*.

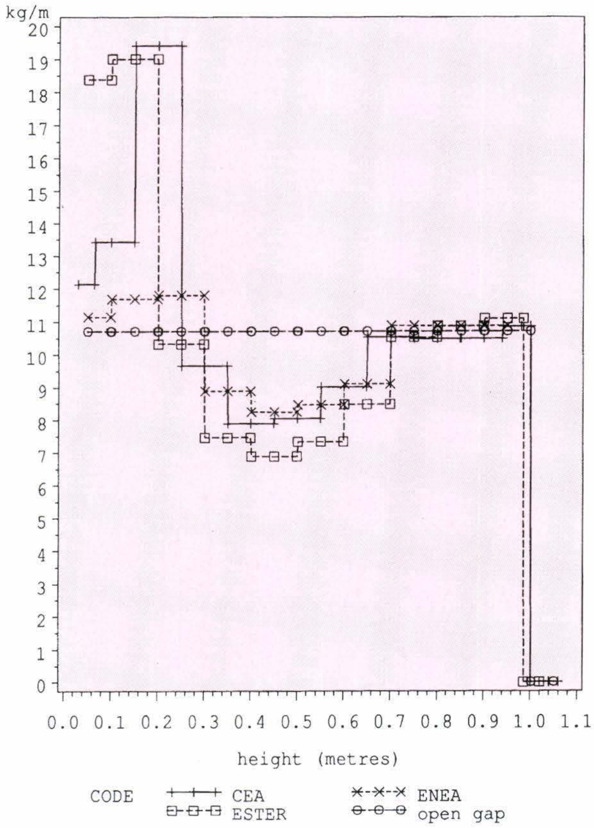

Fig. 1.9 Mass of UO_2 16000 seconds FPT0

Circuit calculations for FPT-0

THERMALHYDRAULICS

It has long been realized that the bundle code ICARE is capable of calculating the thermalhydraulics of the Phebus FPT-0 circuit and indeed any once-through flow path where there are no loops or branches. The coupling of the fission product transport VICTORIA code to ICARE in 1992 as part of the ESTER development programme meant that using ICARE for the circuit thermalhydraulics calculations made even more sense.

The results were then checked against the CATHARE calculations performed by CEA and the results found to be more or less identical. *Fig. 1.10* shows the temperature in the steam generator tube 10000 seconds into the transient plotted against distance from the top of the bundle as calculated by ICARE and CATHARE.

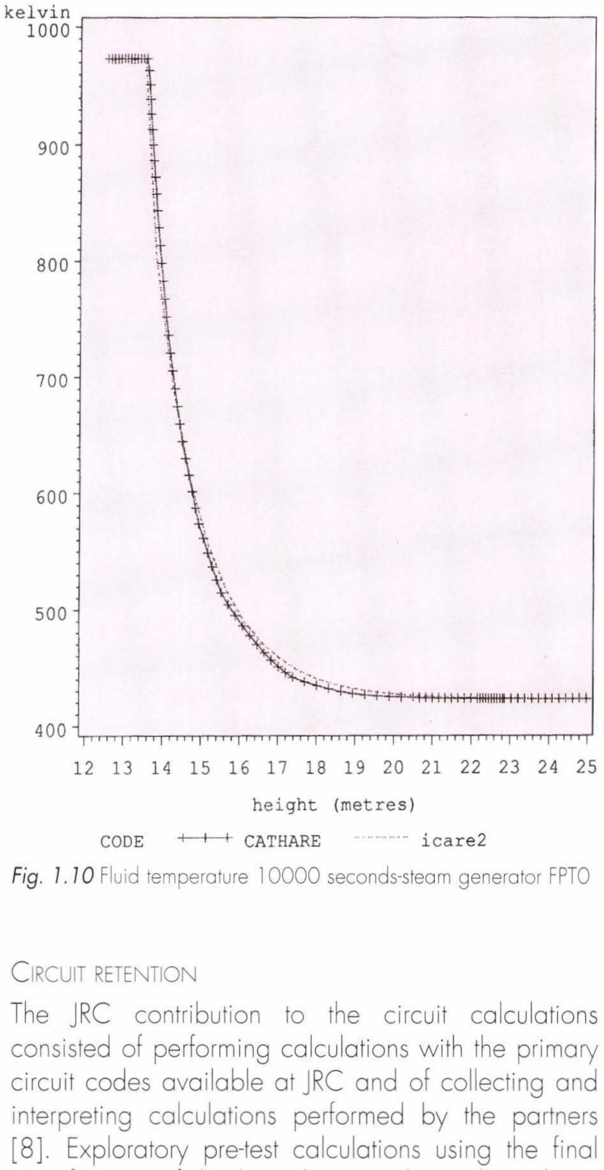

Fig. 1.10 Fluid temperature 10000 seconds-steam generator FPTO

CIRCUIT RETENTION

The JRC contribution to the circuit calculations consisted of performing calculations with the primary circuit codes available at JRC and of collecting and interpreting calculations performed by the partners [8]. Exploratory pre-test calculations using the final specification of the boundary conditions have been performed with the following codes: TRAPF (IPSN version of TRAPMELT-2) at CEN Cadarache, RAFT and VICTORIA (version of May 1991) at JRC Ispra, VICTORIA (version 1992) at Sandia National Laboratories, MACRES at NUPEC Tokyo and ATHLET-CD at GRS Munich. ***Figs. 1.11*** to ***1.12*** show the predicted iodine and tellurium retention in the various circuit compounds.

Both VICTORIA calculations at JRC and at Sandia Laboratories were performed with input data provided by JRC. Even though the VICTORIA versions were different, the final results were in close agreement. The calculations were performed using a 30-cell nodalization of the circuit and with input data provided by CATHARE and ICARE2. Predicted

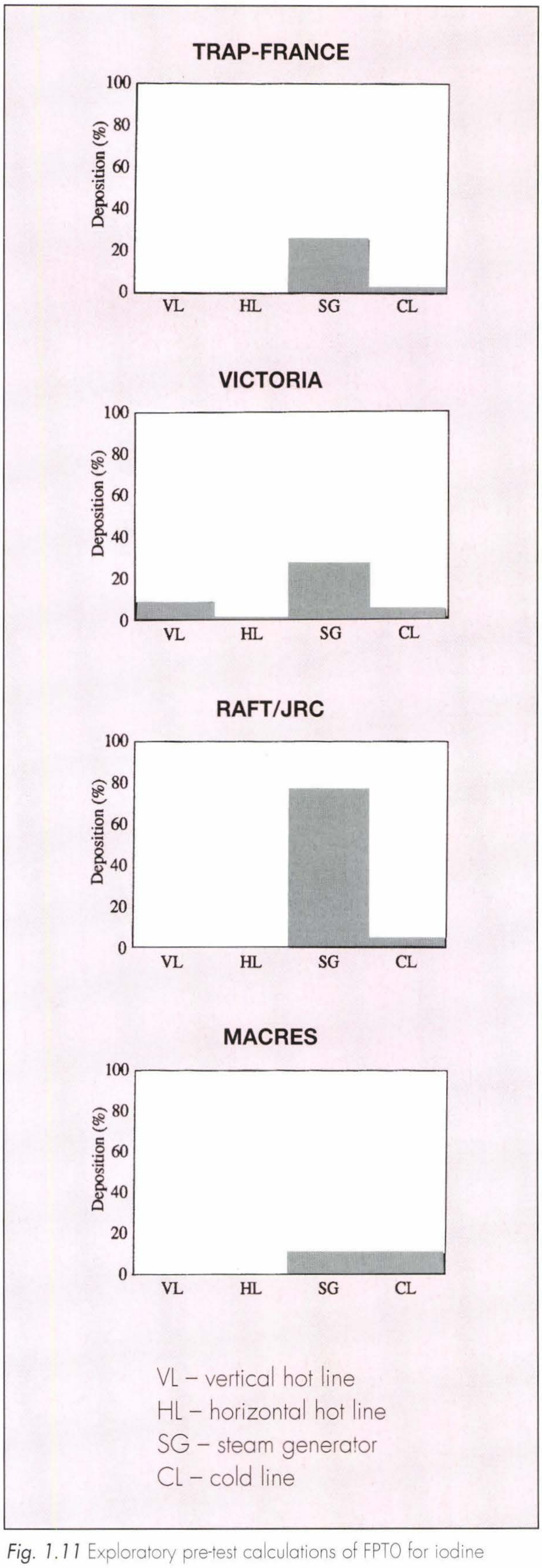

Fig. 1.11 Exploratory pre-test calculations of FPTO for iodine

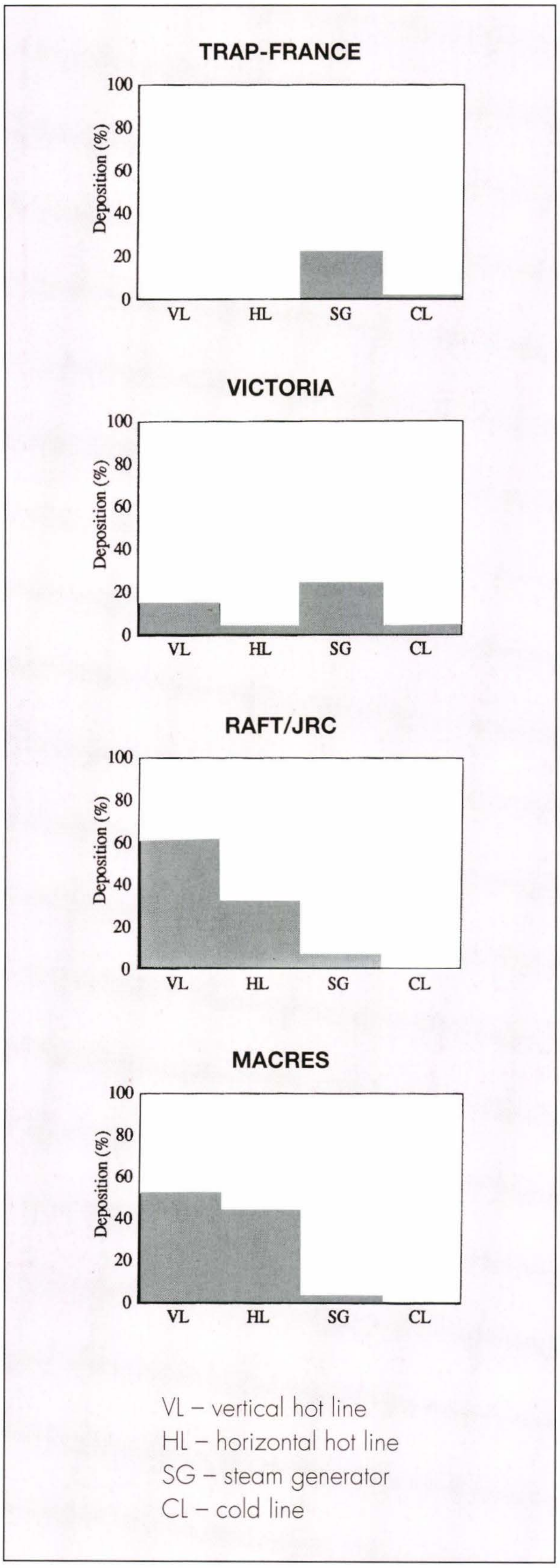

Fig. 1.12 Exploratory pre-test calculations of FPTO for tellurium

deposition (in percentage of injected source) along the circuit for most elements varied between 40% and 50%. Specifically, caesium deposition was predicted to be 40%, iodine 42%, tellurium 47%, barium 42%, silver 40%, and uranium (aerosol) 42%. As shown in *Fig. 1.13* the region of highest deposition is the entrance of the steam generator where the main deposition mechanism is predicted to be thermophoresis. Gravitational settling is the main deposition mechanism before the steam generator, and bend impaction in the cold line. The fission products are released in the containment in aerosol form: iodine as CsI, caesium as CsOH and CsI, and tellurium as Te.

The VICTORIA calculations showed that some elements, most notably iodine, that initially deposit as aerosols (CsI) at the entrance of the primary circuit revaporize at later stages of the transient. The revaporization of iodine from the heated part of the upper plenum (*Fig. 1.14*) results in an increase, by approximately four orders of magnitude, of the amount of hydrogen iodide that is released in the containment (*Fig. 1.15*). The revaporization is a result of complex chemical interactions that arise when the bundle transient is terminated and the bundle power is decreased to zero (at 16000 sec.). At that time the hydrogen concentration in the carrier gas decreases and hydrogen iodide is produced. Consequently, it is suggested that experimental measurements in the circuit continue for at least 2000 sec after the end of the bundle transient.

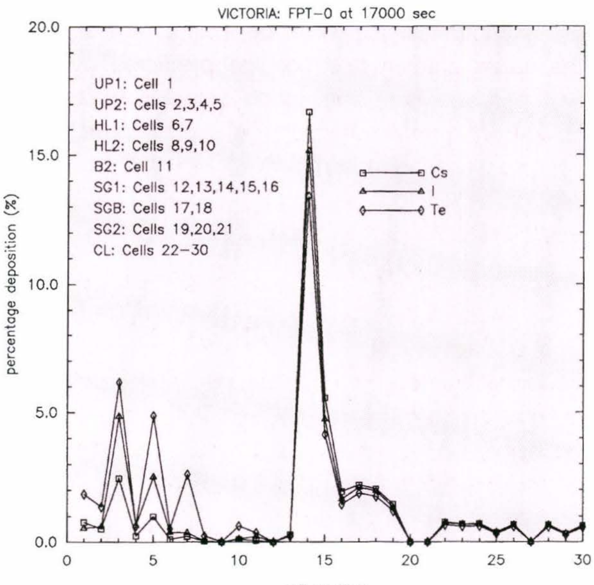

Fig. 1.13 Fission product percentage deposition

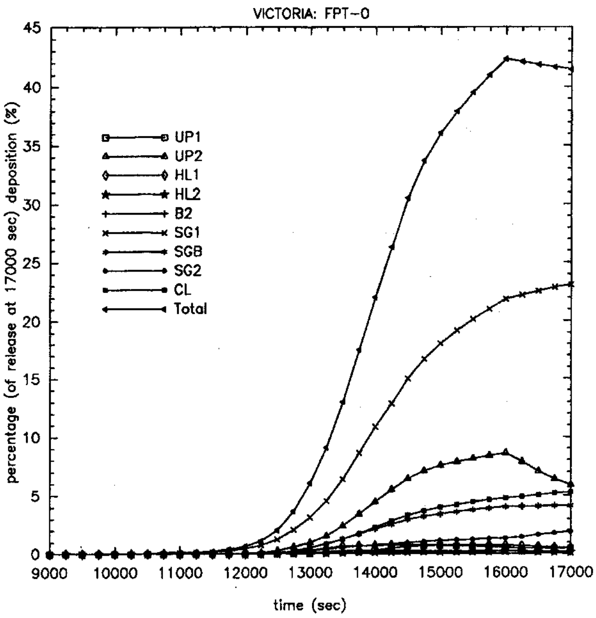

Fig. 1.14 Iodine time-dependent percentage deposition

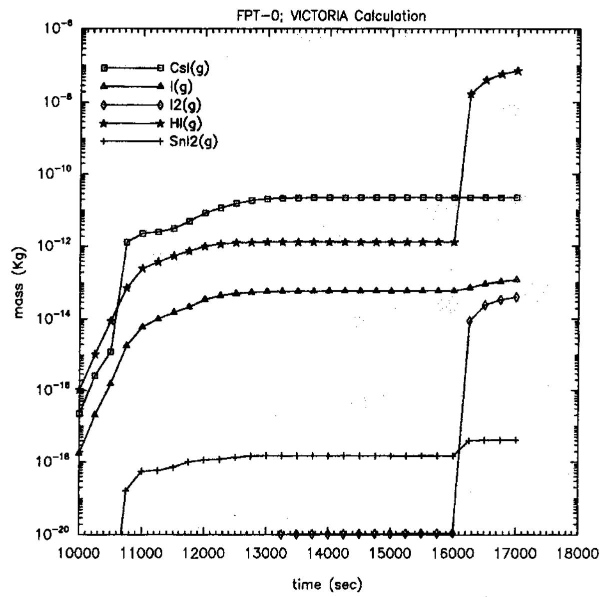

Fig. 1.15 Dominant gaseous iodine species in the containment

Apart from iodine other elements, e.g. tellurium and indium, show non-monotonic deposition profiles. The predicted revaporization is a consequence of changes in the carrier gas composition and in the release pattern. Revaporization, therefore, is affected by the chemistry involved.

The calculations performed with RAFT yielded considerably higher retention values, with 82% of the injected iodine, 86% of the injected caesium and all the injected tellurium being retained in the primary circuit. The main trends, however, are similar, with the exception of the large deposition of tellurium in the upper plenum and hot line predicted by RAFT, which is due to chemisorption between the tellurium vapour and the stainless steel in the pipe walls. Since chemisorption is not modelled in VICTORIA, it predicts very low tellurium deposition before the entrance of the steam generator.

Additional calculations were performed with RAFT using a lower steam flow rate (0.5 g/s instead of 1.5 g/s), which showed a large decrease in vapour condensation onto the steam generator walls, and an increase of the importance, in terms of circuit retention, of the upper plenum and hot vertical line, where the thermal-hydraulic conditions are not very representative, and where RAFT had considerable difficulties in handling very high supersaturation rates due to the large temperature gradient along the pipe at the top of the fuel bundle.

Differences in the code prediction between VICTORIA and RAFT [9] and between those two codes and TRAPF [10] have been investigated. Besides the tellurium-stainless steel chemisorption model, they arise from differences in the modelling of wall condensation and thermophoresis.

Containment calculations for FPT-0

Coupled thermalhydraulic-aerosol calculations for FPT-0 have been performed at Ispra using the CONTAIN 1.12 code. The CONTAIN code has been chosen in order to provide an independent check on JERICHO, neither code being validated for Phebus. The FPT-0 calculation includes:

- the preconditioning phase (50000 sec) to reach a steady state;
- the inlet phase (18000 sec), characterised by Twall=383 K, Tsump=363 K, power extraction from the condenser (***Fig. 1.16***), steam injection at 423 K (***Fig. 1.17***), and aerosol injection (soluble CsI and CsOH, and insoluble Te and inert aerosol for a total amount of 0.99 kg;
- the deposition phase.

The results for FPT-0 are shown in ***Figs. 1.18-1.23***. The atmosphere, wall, sump and condenser temperatures are given in ***Fig. 1.19***.

Fig. 1.20 shows that the pressure increases, during the injection period, from 0.2 MPa to 0.235 MPa, and then decreases again. The relative humidity

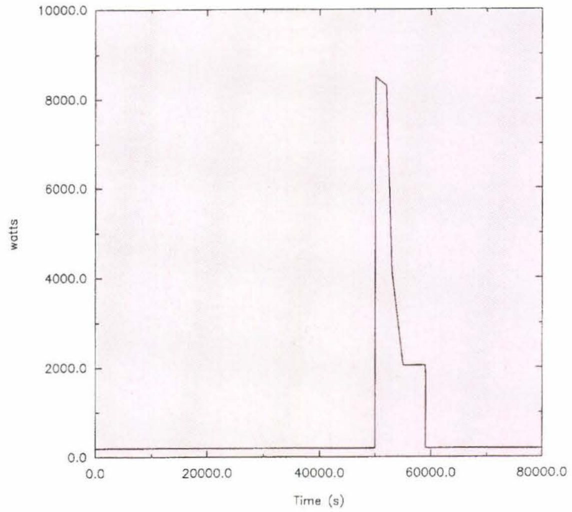

Fig. 1.16 Power subtracted from condenser FPTO

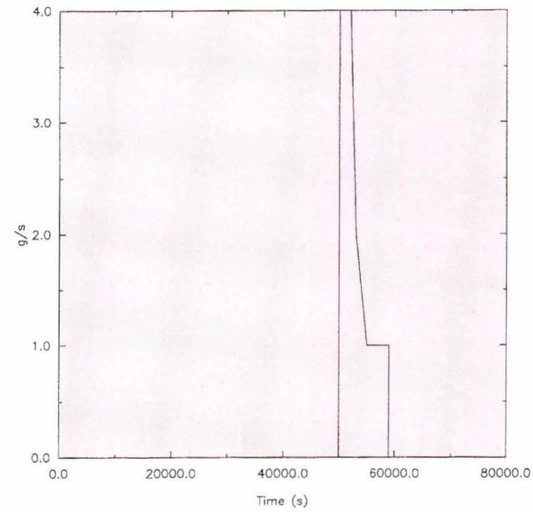

Fig. 1.17 Steam injection rate of FPTO

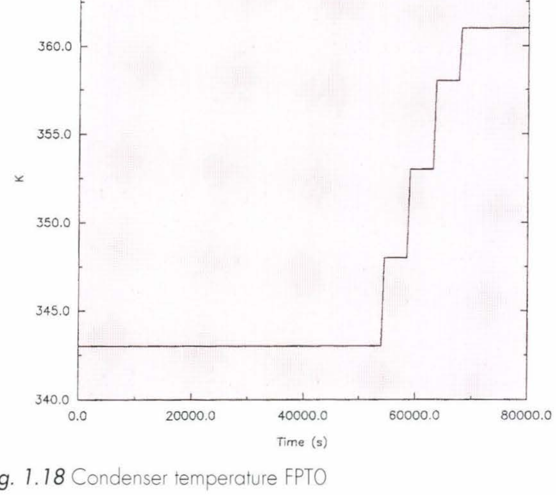

Fig. 1.18 Condenser temperature FPTO

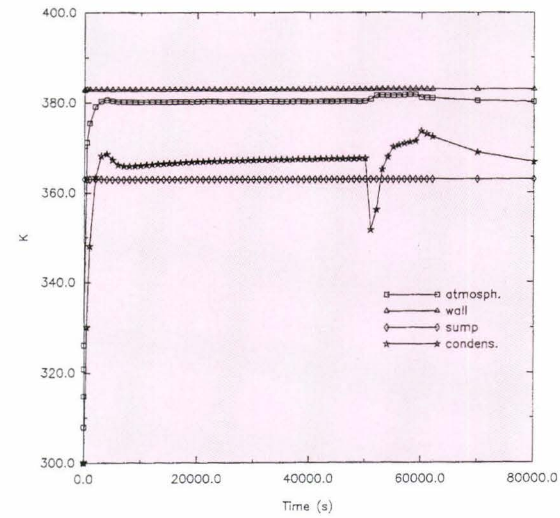

Fig. 1.19 Atmosph. wall sump condens. temperature FPTO

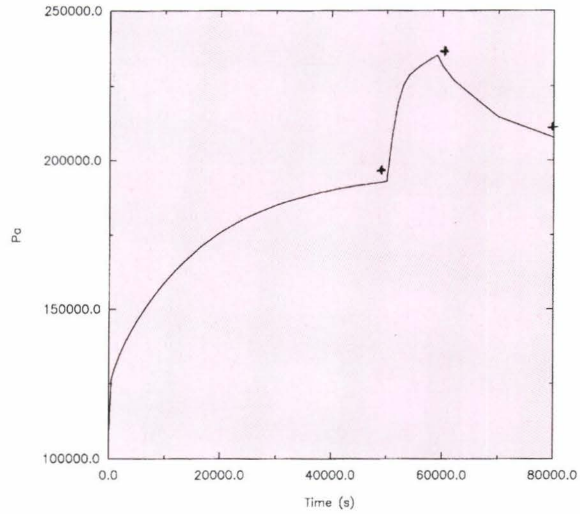

Fig. 1.20 Pressure FPTO

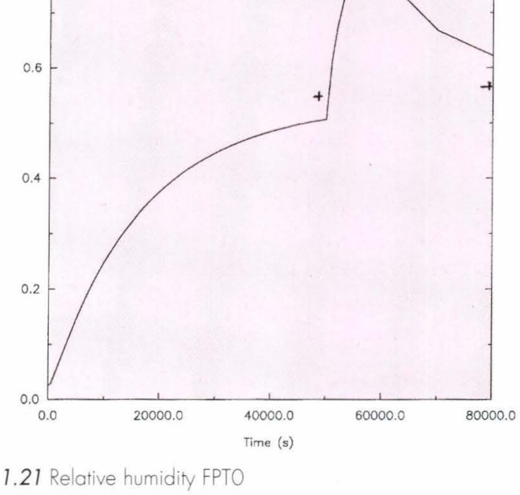

Fig. 1.21 Relative humidity FPTO

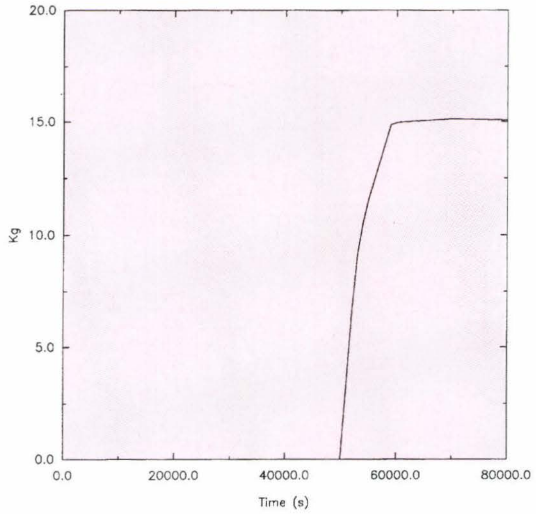

Fig. 1.22 Condensation on condenser FPTO

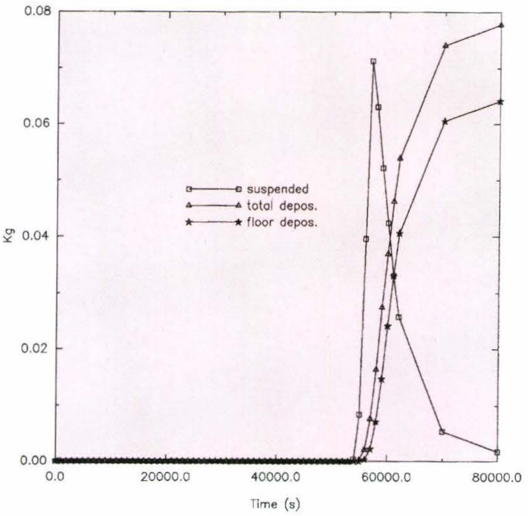

Fig. 1.23 Aerosol airborne mass, deposition and settling in FPTO

(*Fig. 1.21*) reaches about 80% during the injection period and then, at large time (beyond 80000 sec), is stabilized around 55%. Because the vessel wall is unpainted, and therefore untypical, the injected steam should condense only on the condenser. This is in fact the case. About 15 kg of vapour condense on the condenser (*Fig. 1.22*). The comparison with calculations for FPT-0 performed by Layly and Tirini [14] show a rather good agreement for the thermal-hydraulic conditions (crosses in *Figs. 1.19, 1.20* and *1.21* represent CEA results).

Fig. 1.23 shows the aerosol airborne mass and the removal. Removal of the aerosols is largely by diffusiophoresis, in the early stages, but it is later dominated by settling. The ratio of settled to plated mass is in the right range for reactor sequences [11] despite the too low humidity, chosen for technical reasons.

There is similarity between the airborne mass and the total deposition (*Fig. 1.23*) and those obtained by Layly-Tirini (*Fig. 1.35*). The difference is due to the different total amount of aerosol mass assumed to be injected, 99 and 144 g respectively. The ratio between settled and plated aerosols seems to be higher in the CONTAIN calculation. But in this case the total deposition on the floor is considered, which includes also a contribution by thermophoresis, the sump being colder than the atmosphere.

The CONTAIN results are broadly similar to those performed by CEA using JERICHO and this gives us confidence that the proposed boundary conditions will achieve their objective.

FPT-1 Test Preparation

The shroud design employed for the first Phebus test is not wholly satisfactory. The thermal conductivity is too high (resulting in low outlet temperatures unless the steam flow is kept rather high) and its effective value varies rather strongly with temperature due to the gradual closing of the gaps between the two concentric cylinders of zirconia of which it is made. For test FPT-1 a new design of shroud is under consideration, made of fibrous zirconia with an internal coating of dense zirconia to exclude the carrier gas from the fibrous part (where it is known to affect the thermal properties). Some exploratory calculations have been made using estimated geometric and thermal properties of the new shroud, detailed below. The general impression is favourable, the lower effective conductivity resulting in a less strongly peaked axial temperature profile and a higher outlet temperature.

Some predictions of circuit retention have been made with an assumed bundle transient and the higher concentrations of released FP caused by the longer pre-irradiation period (15 days instead of 9 days in FPT-0) and the presence of longer lived fission products resulting from the previous irradiation history of the BR3 fuel rods.

No strong effect of the increased FP concentration is seen but there are several interesting secondary effects. Numerous calculations have been made to explore the different circuit options proposed for this experiment, to prepare the Phebus Committee decision on the topic, and to gauge the effect of boric acid injection into the circuit, which is in the test matrix for FPT-1.

Calculations for the containment portion of the experiment have considered extensions of the thermal-hydraulic tests to the higher humidity conditions planned for the aerosol phase of FPT-1 in which condensation on aerosols is an objective [5]. Calculations have also been performed to predict the expected iodine radiochemistry under the conditions of this test: higher FP concentrations, a consequently higher radiation field and a sump which is initially alkaline (pH=9) and may be buffered.

Bundle calculations for FPT-1

At present the new shroud design has not been finalized. Until it has been the only difference between FPT-1 and FPT-0 as far as the bundle is concerned is that irradiated fuel rather than fresh fuel will be used. This means that fission product release will be different from FPT-0 both because of the different fission product inventory and also because the aged fuel has a different microstructure. Calculating the release is difficult and as a first step a benchmark study is being organized. This is described more fully in the chapter "Code assessment and benchmark problems". The different fuel has little impact on the degradation behaviour of the bundle and none at all for the codes therefore all degradation calculations carried out for FPT-0 are valid also for FPT-1 [12].

Circuit calculations for FPT-1

The JRC contribution to the circuit calculations for test FPT-1 consisted of performing calculations with RAFT to help with the decision on keeping the same geometry as in FPT-0 or replacing it with a minimum retention line, and other calculations with VICTORIA to define what the conditions should be in case the minimum retention line option was selected.

The RAFT calculations were carried out for the full transient and the whole primary circuit, with two possible diameters for the minimum retention line, 20 mm and 40 mm, while with VICTORIA only the part of the circuit downstream from point C was modelled, assuming steady state conditions that were considered representative of the aerosol transport conditions during the transient. The comparison between the FPT-0 geometry and the 40 mm minimum retention line has shown an increase in the amount of caesium and iodine reaching the containment, from 21% and 16% to 52% and 58% respectively. Turbulent deposition and inertial impaction on bends, which play a significant role in the FPT-0 geometry (they are responsible for 36% of the total particle deposition), become negligible for the minimum retention line, where only thermophoretic deposition is important in what concerns aerosol particles.

VICTORIA calculations were performed for the minimum retention line (a horizontal pipe of 4.5 m length) in order to determine the optimum configuration if this option was retained for FPT-1. The pipe diameter (20 mm and 40 mm) and the distance over which the wall temperature dropped from 700°C to 150°C (step transition or over 3 meters) were varied. It was shown that the optimum configuration for average, steady state FPT-1 values was a line of 40 mm diameter with a temperature decrease over 3 meters. No significant dependence on the source rate was noted (FPT-0 or FPT-1), but a strong dependence on aerosol size was observed when the injected aerosol size was large.

As in the case of FPT-0 reasonable agreement between the results of TRAPF (for CEA calculations) and VICTORIA was obtained, and a factor of two difference between those and the RAFT results.

Containment calculations for FPT-1

It is not yet clear how best to specify the boundary conditions for FPT-1. Two scenarios have been investigated:

- With boundary conditions similar to FPT-0 but with the vessel wall temperature at 100°C, instead of 110°C.
- A more radical solution where the vessel wall is cooled below the temperaure of the sump.

The second solution produces higher humidities but does not completely avoid unwanted condensation on the vessel wall.

Scenario I

The FPT-1 calculation performed at Ispra is similar to that for FPT-0 but with the condenser temperature low as shown in *Fig. 1.24*. The atmospheric conditions are near to saturation (*Fig. 1.25*) but no condensation on the walls occurs.

The aerosol airborne mass and the removal are both very similar to the ones in FPT-0. The different conditions, however, influence the role of the different

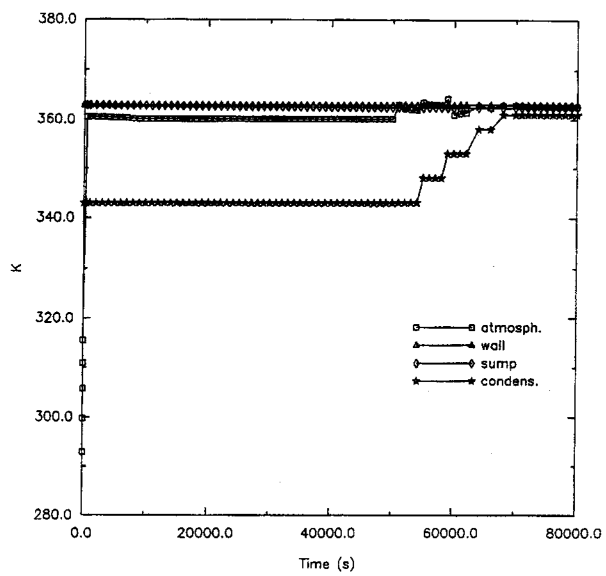

Fig. 1.24 Atmosph. wall sump condens. temperature FPT1

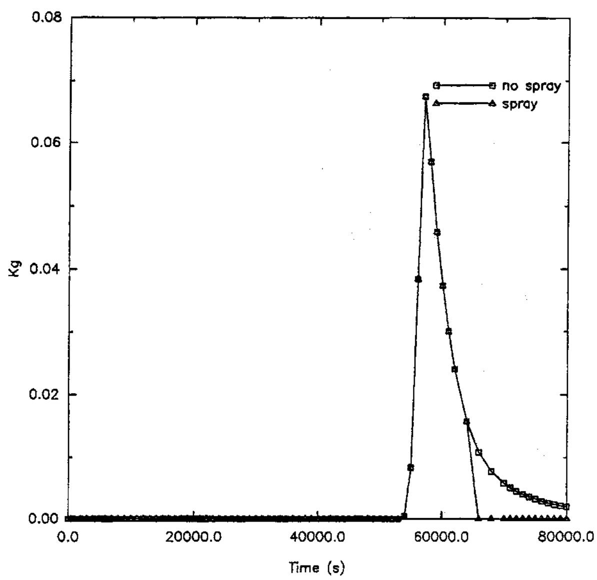

Fig. 1.26 Aerosol airborne mass FPT1

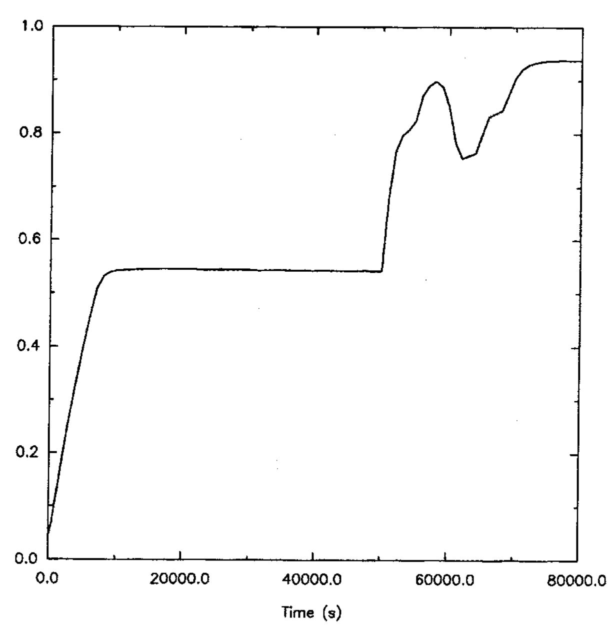

Fig. 1.25 Relative humidity FPT1

mechanisms of removal. In FPT-0 the deposition on the floor is enhanced by thermophoresis, because of the temperature gradient caused by a colder sump. FPT-1 has been considered also with the spray activated at 64000 s [13]. The spray operates at the rate of 0.278 kg/s and temperature of 333 K. The aerosol airborne mass is shown in ***Fig. 1.26***.

Scenario II

Work continues on scoping calculations. The objective is to identify boundary conditions that will obtain the target conditions in the containment vessel (***Fig. 1.27***). In particular the relative humidity should be high during the period that the aerosols are in the atmosphere; i.e. during the injection phase and for two or three hours afterwards. There are also tight technical constraints. The sump should not be more than 90°C and condensation should be mostly on the condenser because it is painted and therefore more representative of a full-scale reactor than the inner surface of the vessel wall which is not.

JRC worked in tight collaboration with the partner organisations. A matrix of calculations was specified that investigated the influence of various parameters, i.e. surface temperatures, steam injection rates and condenser power extraction rates, on a base scenario. The base scenario was chosen to ensure that evaporation from the sump encourages high humidities even when there was no injection and a series of supporting calculations were specified that would investigate the sensitivity to vessel wall temperature, inlet injection rate and the condenser power. The analysis was performed by AEA Winfrith, CIEMAT Madrid and NUPEC Japan using codes that are currently used for reactor calculations (CONTAIN, CONTEMPT, MELCOR) and by JRC using CONT which was specially written for Phebus. The results described fully in [15] and summarized in [5], although differing in detail, did show that high

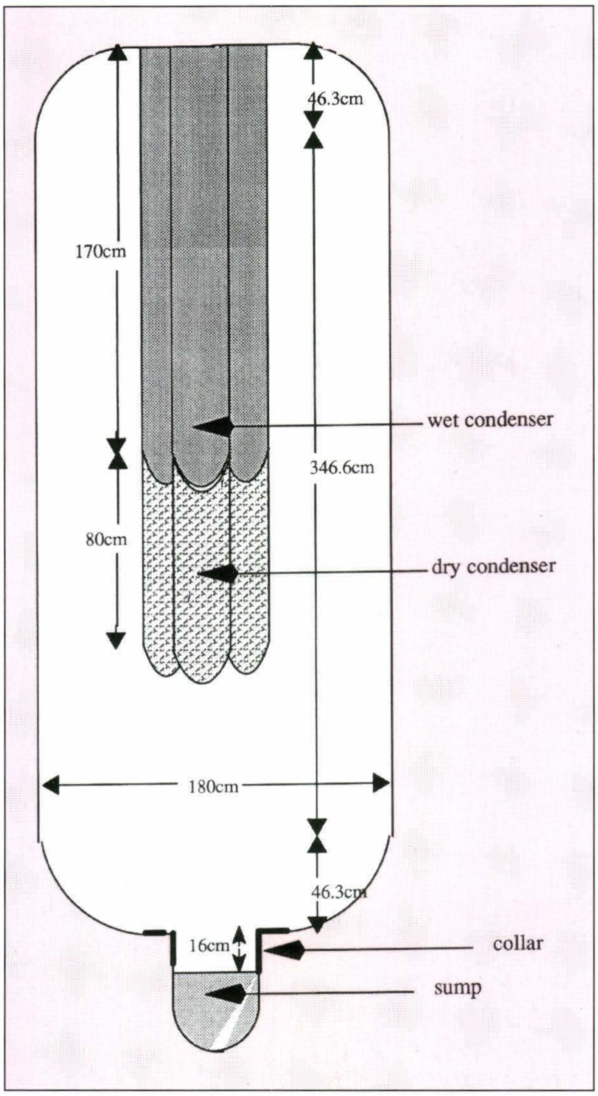

Fig. 1.27 Target conditions in containment vessel

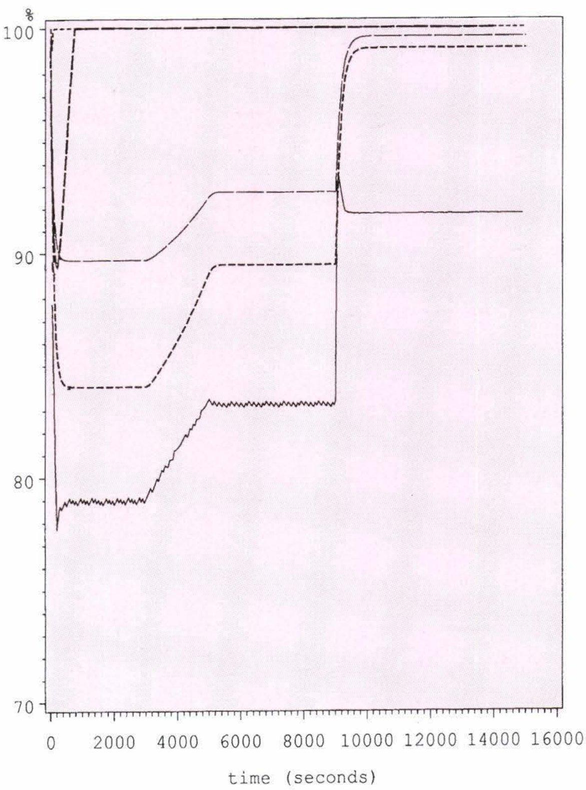

Fig. 1.28 Humidity FPT1

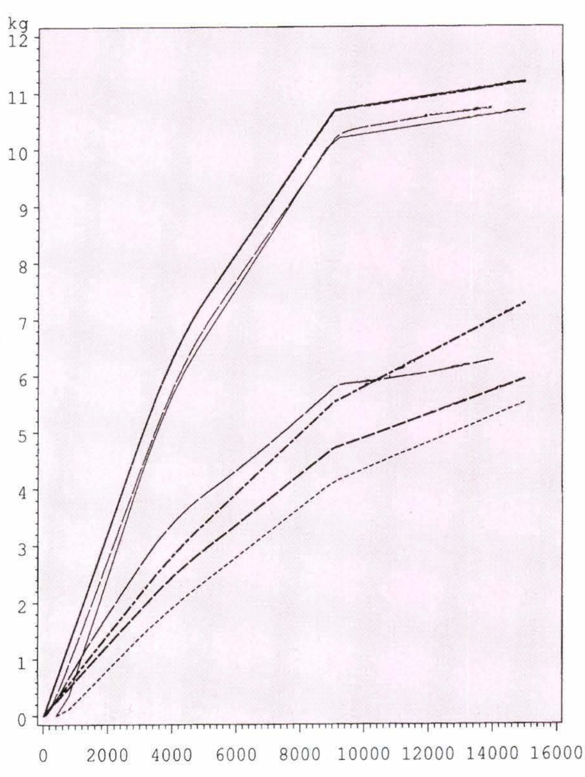

Fig. 1.29 Condensation on wall and condenser FPT1

humidity ratios were possible and that, although undesirable condensation was obtained on the inner surface of the vessel wall, it was always less than the condensation on the condenser. *Figs. 1.28* and *1.29* show the humidity calculations and the integrated condensation rate.

Thermalhydraulic Testing of Containment Vessel

The spread in results in containment thermal-hydraulics calculations was larger than had been expected and, because experimental data are lacking for condensation phenomena in such conditions, there are no a-priori grounds for preferring the results of any paricular code. The difficulties in defining the correct boundary conditions were then compounded when the experimental team realized that they were not able to guarantee a specified power extraction rate on the condenser and much preferred a specified surface temperature. Originally there were plans to build a full size mock-up of the Phebus containment to help understand the thermal-hydraulics better. On grounds of cost it was decided to perform these tests in the actual Phebus vessel before the first test. Both low humidity test, typical of FPT-0, and high humidity test, typical of FPT-1, will be performed. JRC worked closely with the CEA experimental team to define a set of tests and, together with CIEMAT and NUPEC, and using the same codes as before, performed a set of scoping calculations for these tests. Each case is run with the sump at 90°C, the vessel wall temperature and inlet steam flow at constant values and the condenser temperature varied in a series of plateaux.

Figs. 1.30, 1.31 and *1.32* above show the humidity, temperature and saturation temperature for the case where the vessel wall was at 100°C and the inlet flow rate 1.5 grams per second. They indicate that the major uncertainty in the calculation of humidity is the atmosphere which in turn depends mostly on the sensible heat transfer.

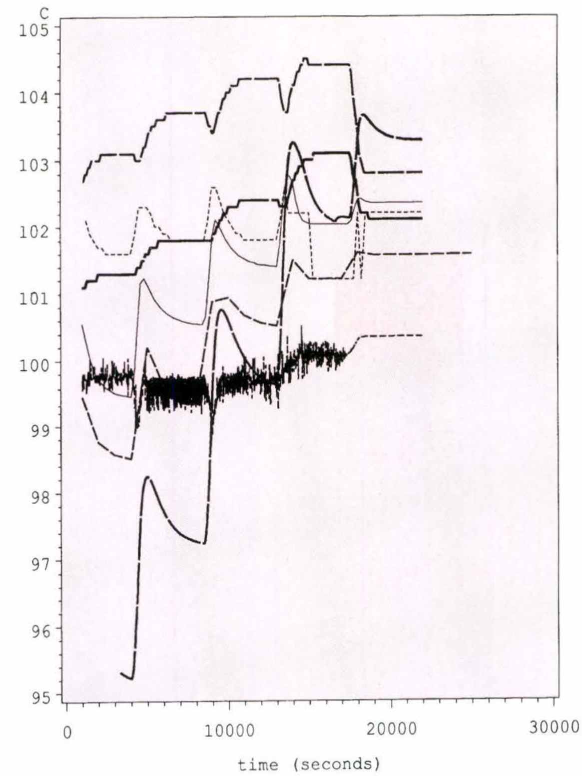

Fig. 1.31 Thermal-hydraulics testing, case 2: atmosphere temperature

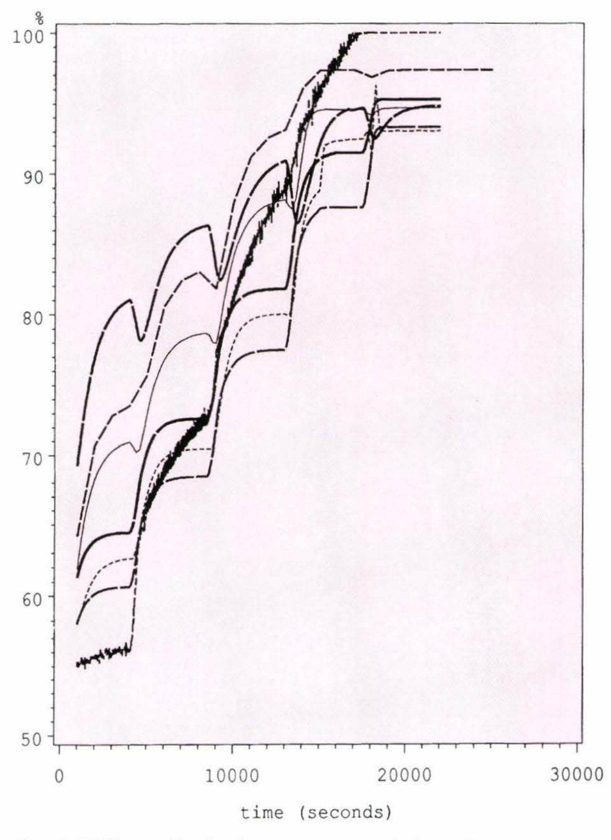

Fig. 1.30 Thermal-hydraulics testing, case 2: humidity

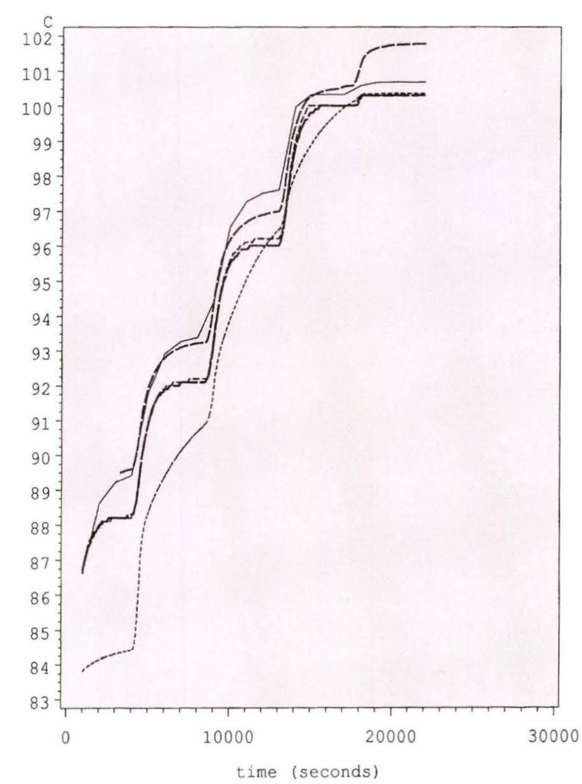

Fig. 1.32 Thermal-hydraulics testing, case 2: saturation temperature

Exploratory Calculations for FPT-2 and Beyond

An outline test protocol has been written for test FPT-2, as a basis for exploratory calculations [16]. The circuit is assumed to be the minimum line configuration which was one option for FPT-1, and the containment conditions are high humidity during the aerosol phase and an initially neutral sump (pH=7) without buffering additives in order to investigate the evolution of the pH as a result of radiolysis (including radiolysis of the dissolved air), hydrolysis of fission products, control material and fuel etc. The conditions for the bundle (according to the existing test matrix) are specified to be reducing during the period of FP release. In the draft scenario the usual calibration plateaux, under pure steam injection are followed by a higher temperature transient, again with steam injection, in order to obtain a representative oxidation of the cladding. This is then followed by a longer high-temperature transient with extensive clad melting and eutectic formation in a carrier gas of 90% helium and 10% hydrogen by volume. Pure hydrogen cannot be used because of safety restrictions on the final quantity of hydrogen in the containment vessel.

Calculations have been made with ICARE to investigate the influence on the core degradation behaviour of:

- Using a hydrogen/helium coolant instead of steam (the FPT-2 scenario).
- Operating at a higher pressure.

These calculations are summarized in [7] and indicate that, according to ICARE, it is possible to reach similar fuel temperatures to FPT-0 with either of these scenarios. They also indicate that the influence of the system pressure on the degradation is slight but significant. Substitution of steam coolant for a hydrogen/helium mixture will, on the other hand, lead to early fuel dissolution by molten zircaloy. *Figs. 1.33* and *1.34* below show the percentage of clad oxidized as a function of axial position at 0.2 MPa, 3.5 MPa and 10 MPa and the mass of uranium dioxide as a function of axial position at 14000 seconds both for the base case (FPT-0) and for the case with a hydrogen/helium mixture. One comforting result from these calculations was that the results at 3 MPa were very similar to those at 10. This is reassuring because it means that, according to ICARE, the maximum technically possible pressure in Phebus (≈3.5 MPa) can simulate transients where the pressure is much higher.

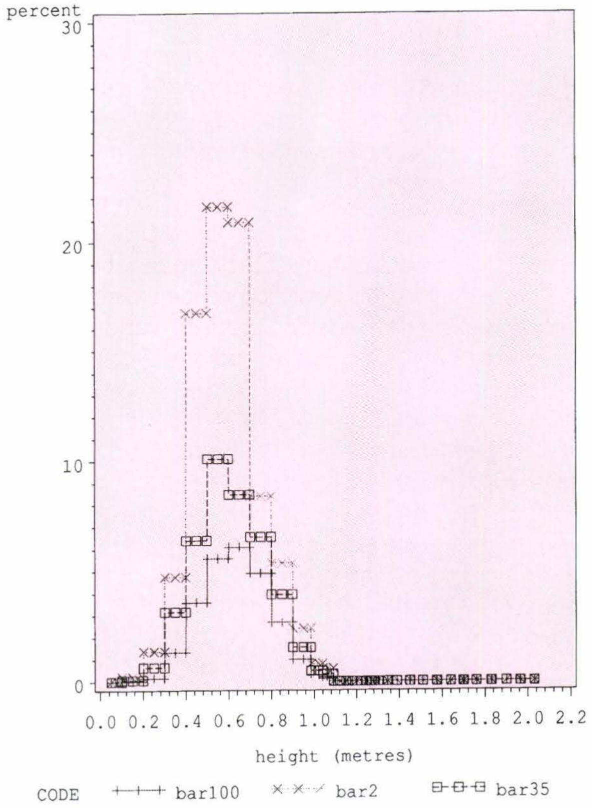

Fig. 1.33 Percentage of ZrO_2 10000 seconds FPTO

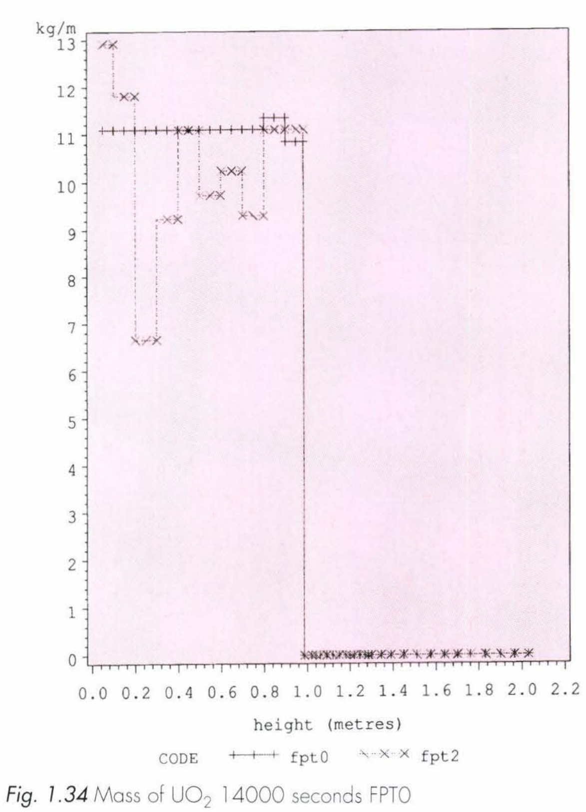

Fig. 1.34 Mass of UO_2 14000 seconds FPTO

Code Developments

Modelling studies - results of Shared Cost Actions (SCA)

Most of the modelling work done under SCAs has been completed, the 1992 crop of SCAs being largely devoted to the integration of several essential or alternative modules in the ESTER code. The model development work this year has concentrated on the interaction between fission product vapours and aerosol particles, both in the primary circuit and in the containment. The Polytechnic University of Madrid (UPM) has under SCA contract developed a theory of the interaction of tellurium, caesium and iodine vapours with aerosols of control rod material (silver, cadmium), and also for the interaction of tellurium with steel surfaces. The theory models the diffusion of e.g. tellurium through the growing layer of reaction product, e.g. silver telluride, on the surface of the aerosol particle. This diffusion becomes the rate-controlling step after a short initial period when the reaction rate is controlled by mass transport in the vapour phase. Hence parabolic reaction kinetics are predicted, the governing rate constants of which are deduced from fundamental transport properties of the product layer via the theory of Wagner, originally developed to describe corrosion processes. The models developed by UPM for tellurium transport have been implemented in a modified version of the RAFT code [17], which has been tested against some of the Marviken experiments. One feature of these experiments is that the deposition profile of tellurium is similar to that of other elements, while codes usually predicted more tellurium deposition upstream due to chemisorption. The modified RAFT predictions come closer to experiment, since according to the calculations the tellurium reacts predominantly with the silver and other aerosols rather than with the steel structure, and so is co-deposited with them.

VICTORIA

Model development and improvement of the numerics in the VICTORIA computer code continued in 1992 under a Shared Cost Action placed with AEA Technology, U.K. [18]. The aerosol model development included a model for aerosol deposition in complex geometries (abrupt pipe contractions, steam separators, and steam driers), while the chemistry models were expanded to include heterogeneous chemical reactions (chemisorption of CsOH on stainless steel) and clad chemistry. Significant improvements were introduced in the numerics. The vapour and aerosol transport routines were rewritten and they were made fully implicit, while an arbitrary cut-off in the chemical equilibrium calculation was eliminated. As a result of the analysis of Falcon experiments (UK), the boundary layer determination and the calculation of the diffusion coefficients were modified to obtain better agreement with experimental data. A new model for control rod release was included, and the assessment and expansion of the VICTORIA thermodynamic data base was completed [19]. Many of these improvements were incorporated in the VICTORIA version that was integrated in the ESTER architecture [20].

The treatment of multi-component aerosols is a notoriously difficult problem. Two different approaches were used to improve the multi-component aerosol treatment in VICTORIA. At AEA Technology Culcheth [21] the coupling of the chemistry module to the aerosol module was modified to render it consistent with the multi-component approach. This new scheme is still under development and validation. At the Polytechnic University of Madrid [22] a separate treatment of nucleation and condensation on aerosol has been attempted on the basis of the already existing multi-component aerosol treatment in VICTORIA.

INSPECT

The INSPECT code, which was initially developed as a mechanistic code for the description of iodine chemistry in the containment, has been expanded to include vapour-aerosol chemical interactions involving both aqueous and non-aqueous aerosols [23]. The model was based on experimental data for iodine vapour interaction with silver aerosols. According to the experimental data the reaction rate may be either linear (mass transfer limited) or parabolic (limited by diffusion through a product layer). Both models were incorporated in the code.

The resulting model was partially validated against ACE CSTF experiments, where CsOH and MnO aerosols were suspended in a large containment. The analysis of some Falcon and ACE/RTF experiments showed that, in general, there was good agreement between code predictions and experimental data.

INSPECT was extensively rewritten and made compatible with the ESTER architecture. It was modified so that it can use the ESTER/RSYGAL library routines to pass information to the ESTER database.

ESTER Model Implementation

ESTER makes available a general framework and a set of computational facilities within which modules from various sources (including the JRC) can exchange data and be intercompared so as to perform complete or partial severe accident calculations. A general description of ESTER has been published in several papers, most recently in [20]. The underlying objectives of ESTER are to enhance European collaboration upon source term code development and to increase the overall quality of codes and modules describing source term phenomena. Based largely on shared-cost action contract work the following status of ESTER has been reached: the database structure has been defined, and a full range of tools for database manipulation are available. These include input reading and checking facilities, graphics and other output facilities, and a menu-based user interface. A first preliminary version of ESTER has been issued to volunteer organisations for portability and other tests. It contains only two physical molecules, ICARE-2V2 from CEA, and FPRATE, a fission product release model from the German core degradation code system KESS. As described below, a number of additional physics and chemistry modules are being integrated into ESTER under SCAs, and as they come in, the JRC is attempting to complete the integration by the creation of appropriate drivers etc. in order to apply the result to Phebus tests, the International Standard problem and other problems of immediate interest, for validation purposes.

To date the ICARE + VICTORIA combination has been used to calculate the Falcon ISP using ICARE to calculate the thermal-hydraulics in the circuit, and ESTER/JERICHO has been applied to the thermal-hydraulic tests planned for Phebus.

Other modules currently being implemented include the essentials of the KESS core degradation code system from IKE Stuttgart, the GRS containment codes FIPLOC and RALOC, the pool scrubbing code BUSCA, and the core-concrete interaction module WECHSL. A further module being integrated is the iodine chemistry code INSPECT, enhanced by a new model for aerosol-vapour interaction suitable for the containment.

Further SCA work has concerned the feasibility of the integration of the German system thermal-hydraulics code ATHLET into ESTER (GRS+IKE). A recommendation of this study is that one should strive for a loose coupling between the codes. The core region could be treated by a two-phase thermal-hydraulics module under development by CISI, while ATHLET could handle the remainder of the vessel and the reactor circuit with its multiple junctions, components etc.

The structure of ESTER has been further extended by the addition of facilities (IKE) for remote procedure calls. These enable different modules of ESTER to be invoked on different computers. For instance a relatively slow-running module could be handled by a CRAY, while on-line or off-line graphics could be confided to a specialised graphics work-station. The same tools can allow parallel operation if and when the logic of ESTER so permits, bearing in mind constraints of data access and overall logic.

VICTORIA-ESTER Coupling

A standalone version of VICTORIA had been integrated into ESTER by AEE Winfrith, UK. At JRC this version of VICTORIA was coupled with the thermalhydraulic code ICARE. The ESTER version of VICTORIA contained modules for calculating aerosol and vapour transport in the coolant but did not include the original VICTORIA's fuel modelling because ICARE's CORSOR model was considered adequate for the time being. The flow of information between ICARE and VICTORIA is shown in *Fig. 1.35*. Normally VICTORIA is nodalized much more coarsely than ICARE because, if it were nodalized any finer, it would be run too slowly to be useful.

This coupling makes VICTORIA a much more useful tool. Simplified input preparation and consistent thermalhydraulics reduce errors. Test cases for Phebus and Falcon have been run and are described in another section of this report. The code has been sent to the Universities of Stuttgart and Dresden where it has been implemented on Unix workstations and is being enthusiastically applied to the modelling of fission product transport.

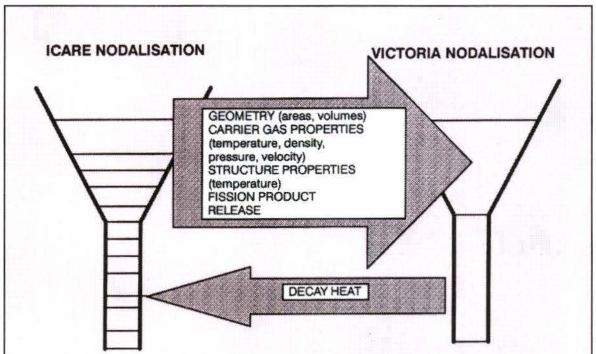

Fig. 1.35 Flow of information between ICARE and VICTORIA

ESTER Installation and User's Club

The preliminary version of ESTER containing the bundle degradation ICARE2 and the fission product release FPRATE modules, together with the RSYGAL tools, was delivered by the SCA contractor CISI Ingenierie, France to the JRC in the beginning of 1992.

A first ESTER Workshop was organised end of June to promote and to identify the potential users of the programme. An ESTER User's Club was officially constituted. Following the success of the workshop, the preliminary version of ESTER was sent to ten different organisations, not only academic organisations but also research centres and private companies. Countries concerned are Italy, France, UK, Germany, Spain, Denmark and Belgium.

The ESTER software has been installed on different computers in Ispra, in order to test its portability and to be able to give an external support. Specific procedural, compiler and language problems were identified and resolved. At the end of the year the installations, including some graphical tests, were successfully completed on SUN, IBM and HP computers. Installations on other computers are still underway.

A preliminary version of the coupled ICARE2-VICTORIA ESTER software and a JERICHO ESTER tape were also distributed to different organisations.

Several organizations are presently developing and adapting source term codes and modules to integrate in the context of the ESTER framework. To assess the software quality of these modules, a specific tool FCOMLIS was developed by AIB Vinçotte Nuclear in Belgium. The version 2.07 of the software has been released and distributed to the different partners.

THE PRELIMINARY VERSION OF ESTER HAS BEEN DISTRIBUTED TO THE FOLLOWING ORGANISATIONS:

FZR, Institut für Radiochemie	*Dresden-Germany*
IKE, Institut für Kerntechnik und Energiesysteme	*Stuttgart-Germany*
Ruhr Universitat, Institut für Energietechnik	*Bochum-Germany*
Universidad Politecnica, Catedra de Tecnologia Nucl.	*Madrid-Spain*
Thermal and Nuclear Research Centre of ENEL	*Milano-Italy*
Università di Pisa, Dipartimento di Costruzioni Meccaniche e Nucleari	*Pisa-Italy*
MATEC	*Milano-Italy*
CEA, Centre d'Etudes	*Cadarache-France*
AEA, Technology Centre	*Winfrith-UK*
AIB Vinçotte Nuclear	*Brussels-Belgium*

An ESTER-JERICHO version was transmitted to:

GRS, Gesellschaft für Anlagen und Reaktorsicherheit	*Garching-Germany*

An ESTER coupled version of ICARE-VICTORIA was transmitted to:

AEA, Technology Centre	*Winfrith-UK*
IKE, Institut für Kerntechnik und Energiesysteme	*Stuttgart-Germany*
FZR, Institut für Radiochemie	*Dresden-Germany*
CEA, Centre d'Etudes Nucléaires	*Cadarache-France*
RISØ National Lab., Nuclear Safety Research Dept.	*Roskilde-Denmark*

ESTER Maintenance and Development Plans

The ESTER User's Club was launched in June 1992, through which the JRC incurred a number of responsibilities, notably to correct errors identified by users (including portability problems arising from the wide variety of machines at the users' laboratories), to keep users supplied with updated versions of ESTER as they become available, and to train users in elementary and more advanced ESTER features.

Here one may identify two groups with somewhat different needs: ESTER users pure and simple, and users who are also interested in developing the code, by incorporating new modules or by modifying existing modules to add new features or correct errors. There are further training requirements for the JRC itself, both in ESTER and in setting up an efficient and user friendly maintenance system.

To assist with all these responsibilities and tasks a contract has been signed with CISI Ingenierie, the main developer of the ESTER toolkit. JRC will be assisted to set up a maintenance system, to prepare and deliver user training, in the correction of user errors and in the extension of ESTER facilities. A further User's Workshop will probably be held during 1993 to demonstrate the extensions to ESTER resulting from the incorporation of new modules. ***Table 1*** shows the expected configuration of ESTER at the end of 1993. At this point ESTER should be capable of calculating all the Phebus tests. It is planned to use 1994 to consolidate the ESTER development through the collection of user's experience and validation against various experiments, and to issue an official version of ESTER, ESTER-V1, towards the end of that year.

Table 1.1 Main ESTER modules: version p2

Systems:	RSYGAL RPCs *{GUI, better graphics, video}*
Core/Bundle:	ICARE-2 KESS modules incl. FPRATE CHIP {ORIGEN, FUTURE, FASTGRASS, *Control release, quench,* *lower head relocation, penetration}*
Circuit:	ICARE-2 CHIP BUSCA VICTORIA *{ATHLET, LOCA ⇒ severe accident)*
Containment:	JERICHO, AEROSOLS RALOC, FIPLOC IODE, INSPECT WECHSL *{DCH, MCCI aerosol,* *models for new cont. designs}*

Codes Assessment and Benchmark Problems

RAFT

The RAFT code is being used for Phebus-FP pre-test calculations by the JRC, the Polytechnical University of Madrid and the Centre d'Etudes de Cadarache, but funding for its development by EPRI, the original developer, has stopped, leading the JRC to take an active role in coordinating the developments performed by several different organisations, including the addition of boron species (VTT Finland), and the modelling of chemical reactions between the tellurium vapours and the control rod material aerosol particles (UPM Spain).

The FPT-0 and FPT-1 pre-test calculations have provided a good basis for code comparisons with VICTORIA and TRAPF, which led to the identification of some important differences between physical models or between the implementation of the models in the different codes. This also led to an effort in assessing RAFT against available experimental results.

The hot tube experiments at the Argonne National Laboratory (ANL) have been used to assess the tellurium chemisorption model. The results of test No. 10 (*Fig. 1.36*), showing that RAFT predicts much more tellurium deposition (by chemisorption) near the entrance of the test tube than was observed in the test, led to the conclusion that the chemisorption rate between tellurium and stainless steel in RAFT is too high, probably due to the fact that, after a reacted layer forms on the stainless steel surface, the chemisorption rate is conditioned by diffusion

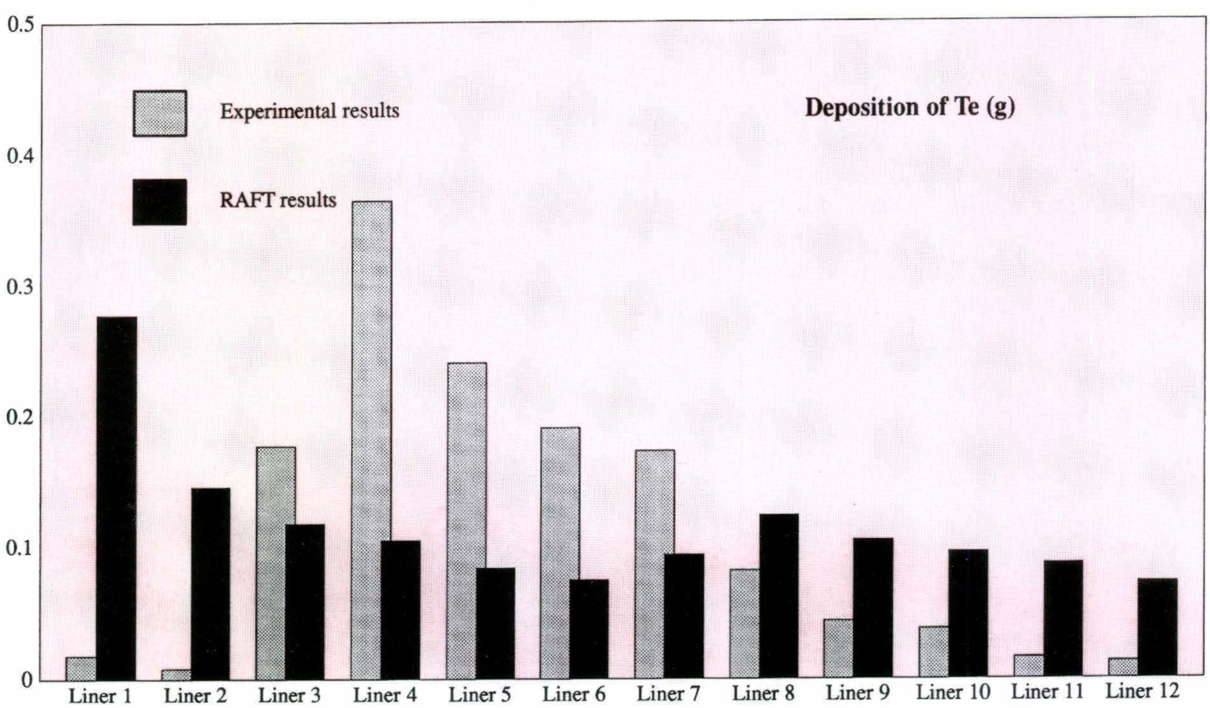

Fig. 1.36 ANL hot tube experiment No. 10

through the reacted layer, becoming parabolic instead of linear. This conclusion will need to be confirmed by other experimental evidence before it is used to modify the chemisorption model in RAFT.

Fission product release benchmark

An international benchmark exercise on fission product release was initiated as a joint effort of JRC/STI and Transuran Inst. (TUI). Participants include organisations from the European Communities, Japan, and the U.S. The objective of the exercise is to compare code predictions for fission product release from fresh fuel (FPT-0) and high burn-up fuel (FPT-1). It addresses one of the weak links in the Phebus-FP pre-test exploratory calculations. The first part of the exercise, which investigated release from fresh fuel, was concluded, and in the coming year the results will be analyzed. The next phase of the exercise, which will be based on release from high burn-up fuel, will begin with the termination of the experimental analysis of the FPT-1 fuel. It is expected that the results of the second phase will be used in the preparation of FPT-1.

Falcon Shared Cost Action Programme and Falcon ISP-34

The Falcon experiments continued under a Shared Cost Action programme placed with AEA Technology, Winfrith (U.K.). Four integral experiments are planned, two of which (FAL-17 and FAL-18) were performed in 1992. Close collaboration between the experimental team and analysts from the U.K., France, and JRC resulted in considerable improvements in data collection and presentation of results. However, uncertainties related to the timing of control rod and fission product release remain. A thermal hydraulic experiment that will resolve uncertainties associated with temperature measurements is planned for February 1993.

The experiments conducted in 1992 were thermal gradient experiments that measured fission product transport and deposition under a thermal gradient that is similar to the thermal gradient at the entrance of the steam generator of FPT-0. Post-test calculations for FAL-17 were conducted with VICTORIA, TRAPF, and VICTORIA91/ESTER (***Fig. 1.37***). The calculations showed reasonable agreement for fission product deposition towards the end of the thermal gradient tube. FAL-18 was performed under identical conditions with FAL-17 except that boric acid was not injected. The experimental results showed that boric acid significantly modified the transport of caesium.

The CSNI-sponsored International Standard Problem No. 34 consists of two Falcon experiments that were performed in 1992. JRC participates in thus international exercises with primary circuit post-test calculations conducted with RAFT and VICTORIA 91/ESTER. The analysis of the results will be completed in 1993.

IODINE-STEEL INTERACTIONS

Under a Shared Cost Action placed with Siemens/ KWU a literature survey of iodine-iron and iodine-stainless steel reactions was conducted. It demonstrated the importance of these interactions. The work is continuing with an assessment of the iodine-steel reactions in the Phebus-FP circuit and containment. The work will conclude in the middle of next year with experiments on the most important iodine-steel reactions for Phebus.

IODINE CHEMISTRY BENCHMARK (ON FPT-0)

Benchmark calculations were performed by the following partners:

CEA, Cadarache	*IODE code*
CIEMAT, Madrid	*IODE code*
GRS,Köln and SIEMENS/KWU, Erlangen	*IMPAIR-2, code*
OH, Toronto; AECL, Whiteshell and University of Toronto	*LIRIC code*
AEA, Harwell	*INSPECT code*

IODE and IMPAIR-2 are empirical codes which use correlations to model the system. INSPECT and LIRIC are mechanistic codes which describe the chemistry in terms of the fundamental reactions which are involved. These codes model chemical reactions in the aqueous and in the gaseous phase, and also mass transfer effects between aqueous and gaseous phases.

Various calculations were performed with/without iodine-surface interaction and with/without organic radicals in the sump water at different sump water pH values.

Code result discrepancies between the two groups of codes (empirical and mechanistic) are acceptable at pH=5 (factor 3) and large at pH=7 (factor 20). Furthermore, INSPECT and LIRIC show considerable agreement even if some rate constants differ.

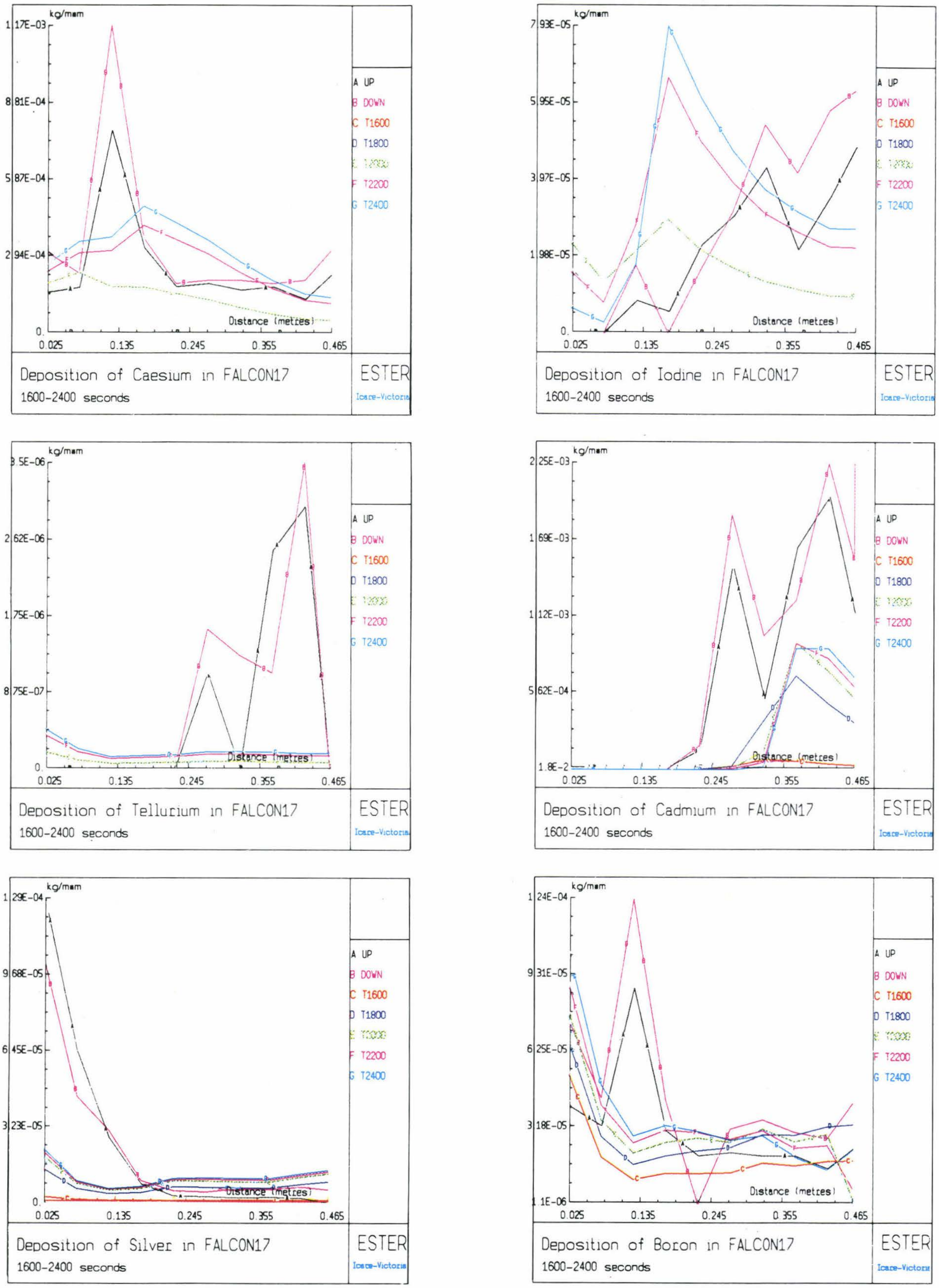

Fig. 1.37 FP-deposition in FAL17

However, IODE and IMPAIR-2 are in disagreement because they model differently HOI disproportions (J.T.Bell formalism in IODE [24] and L.M. TOTH formalism in IMPAIR-2 [25]).

Code benchmark results show also that:

- the amount of molecular iodine in the gaseous phase is independent of the existence of organic radicals in the aqueous phase,
- the molecular iodine concentrations in the gaseous phase are 10 to 100 times higher at acid pH (pH=5) than at neutral pH (pH=7) for sump water,
- the values of iodine concentration in the gaseous phase with a neutral pH (pH=7) are close to the limit of foreseen instrumentation. So it is considered judicious to choose an acid pH (pH=5)* for sump water in FPT-0 in order to maximize the iodine percentage in the gaseous phase, because FPT-0 will be performed with an instrumentation not "run-in" under PHEBUS/FP conditions [26].

STORM

STORM (**S**implified **T**ests **O**n **R**esuspension **M**echanism) is a large-scale experimental facility to study the behaviour of fission products and aerosols in reactor components, see *Fig. 1.38*. The STORM experimental programme is focussed on the investigation of the effect of high gas velocities (< 200 ms^{-1}) on the deposition and resuspension of aerosol particles in pipework [27].

In December 1991 a contract between ENEL (Italian Electricity Board) and JRC has been signed for cooperation in the field of LWR severe accident analysis. In particular, ENEL is providing significant financial support to the STORM project and is contributing to the solution of technical aspects with its personnel.

* pH=5 initially and no buffering substances will be added to maintain the pH constant throughout the experiment (pH may become lower or higher than 5 during the experiment)

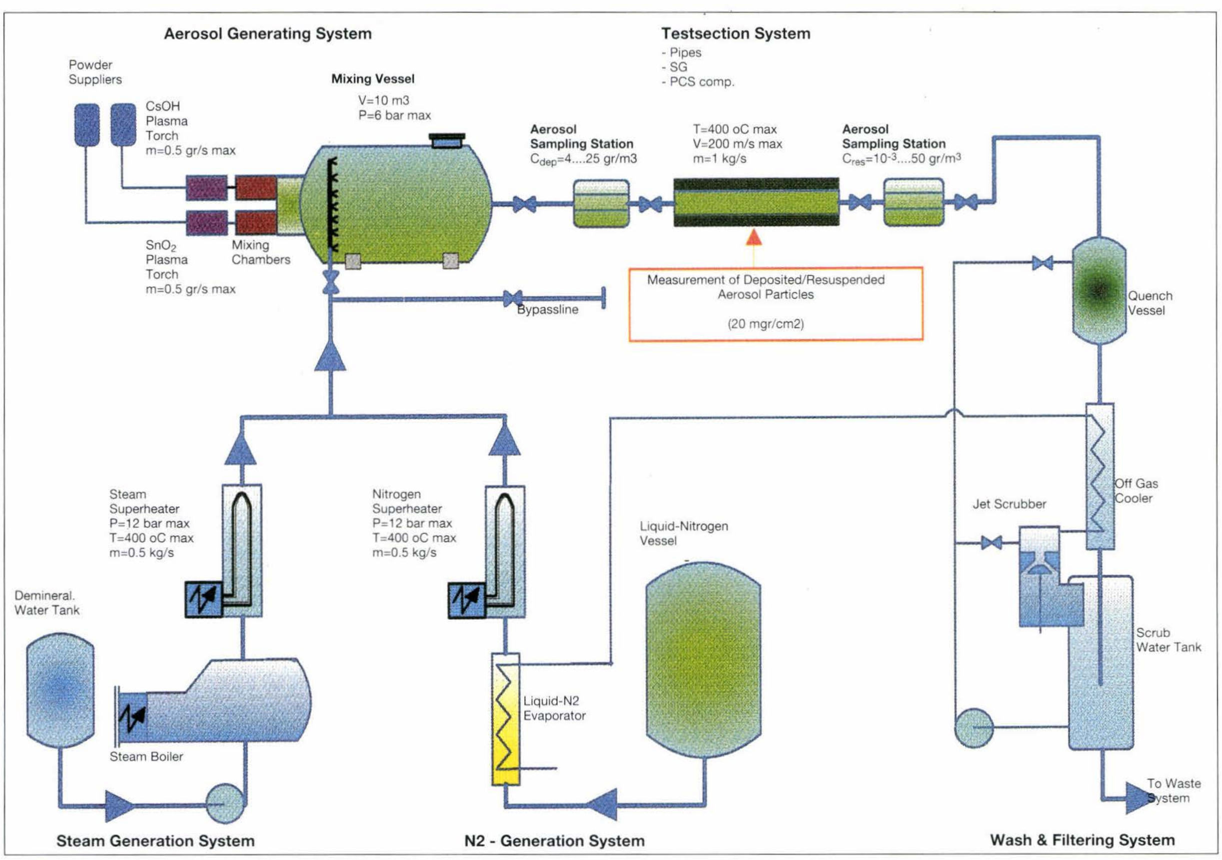

Fig. 1.38 STORM facility

On the basis of the previous aerosol studies, the design of STORM places emphasis on the conditions that are likely to be encountered in the event of a severe nuclear reactor accident (LWR Station Black-Out, PWR V-Sequence, etc.). According to this the experiment will provide a representative experimental data base that can be used for a more complete understanding of the mechanisms of deposition and resuspension of aerosol particles. The development and validation of theoretical or semi-empirical deposition and resuspension models to be included in nuclear aerosol codes are the final target of the programme.

During 1992, the following main activities have been performed:

- review of the reference preliminary design [28] in order to ascertain its feasibility and adequacy and to propose modifications to ensure viability or to secure improvements in performance, safety and costs
- detailed design of the facility, including the engineering analysis of the facility, and sizing of components
- design of the control system
- elaboration of technical specification for component purchase
- procurement of part of the components
- study of advanced measurement techniques
- preliminary analytical code calculations of the reference accident sequences
- preparation of the experimental hall which will host the facility.

Previous Research on Aerosol-Resuspension Mechanism

The overall retention of radionuclides in the reactor coolant system and containment during an LWR severe accident is the result of two concurrent phenomena: deposition and resuspension. While the former comprises the sum of investigated mechanisms included in currently used aerosol codes, the latter is not usually taken into account in present Source Term analyses. However, LACE tests and other experiments of smaller scale have already shown that aerosol resuspension may have relevant influence on the radionuclide retention capabilities of the nuclear island, and that high priority should be given to its modelling in the aerosol transport codes. In spite of the importance of resuspension for a correct Source Term assessment, the present experimental evidence is far from being adequate to quantify the relocation of the material and the reentrainment in airborne form of the previously deposited particles.

A number of empirical correlations have been already developed to estimate soil erosion in agriculture, dust release in mining, and toxic or radioactive powder resuspension from contaminated surfaces [29,30,31]. These studies and correlations, which are of narrow applicability, pointed out that parameters which may influence the resuspension of particles from surfaces include: aerosol nature and size prior to deposition, aerosol deposition mechanisms, concentration of deposits on surfaces, surface roughness, surface moisture conditions. Recent studies in the Source Term field pointed out the peculiarity of aerosol transport throughout the nuclear island, and the consequent need to perform experimental resuspension tests under representative boundary conditions [32,33,34].

Preparatory Code Calculations on STORM Experimental Programme

In order to estimate the importance of resuspension within the primary circuit under LWR severe accident conditions, some typical sequences have been analysed. These sequences are station blackouts in a PWR and in a BWR, and a small LOCA (Loss Of Coolant Accident) in a PWR. All sequences are characterised by the total unavailability of all core cooling systems, with the consequent dry-out of the primary circuit and the core meltdown. Regarding the aerosol transport, the main difference between the two scenarios consists in the discharge into the containment, which is "dry", for the PWR, and through the suppression pool for the BWR. All these sequences have been analysed using ECART (ENEL Code for Analysis of Radionuclide Transport) code, with and without the aerosol resuspension model, which consists of a correlation based on a number of experiments (Oak Ridge, Würenlingen, Winfrith):

$$\lambda(r) = 1.8\ F(r)^{0.88}$$

where

$\lambda(r)$ is the resuspension rate, expressed in s^{-1}

$F(r)$ is the resultant force, expressed in μN, and

r is the deposited particle radius of the current size bin.

The results obtained show that, for each sequence, the effect of resuspension in the primary circuit is low during the core melting period, which is characterised by very low gas flows, while it becomes important after the core slump, when large amounts of steam are produced by debris quenching within the water pool remaining in the bottom of the vessel.

As a consequence, in the case of the PWR scenarios, the amount of aerosols reaching the containment atmosphere is much larger if resuspension is considered (*Fig. 1.39*). Conversely, for BWRs this is true only for the aerosols reaching the suppression pool, so that, compared with the case without resuspension, less material is available for further release from the primary circuit after vessel failure. *Fig. 1.40* illustrates the material relocation within the PWR primary circuit due to resuspension in high-velocity sections and deposition in wider components.

As can be seen, the large surface of the pressurizer heaters traps the aerosols resuspended upstream, while the retention in the low-pressure components (downstream of the pressurizer relief valves) becomes negligible. A sensitivity study has been performed on the threshold value of mass fraction of the liquid phase above which resuspension in inhibited. If this threshold value is low, the effect of resuspension is reduced in the surge line, and mainly in the hot leg, where a liquid phase is more likely to occur.

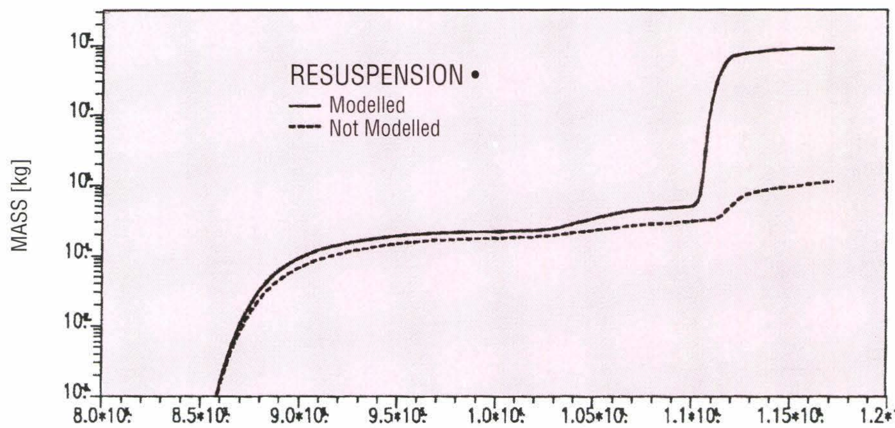

Fig. 1.39 Mass of aerosol particles injected into the containment during the PWR station blackout scenario, as calculated by the ECART code

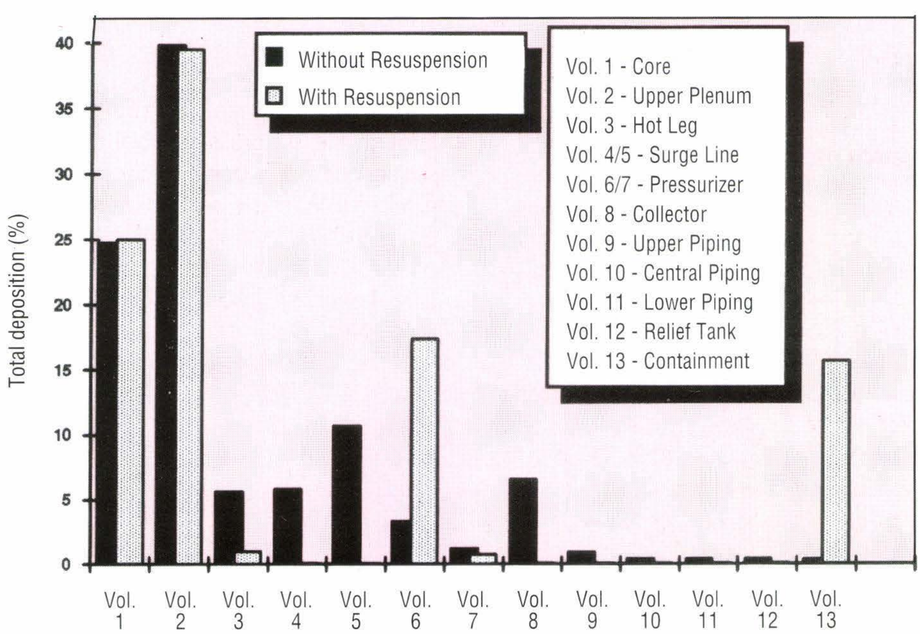

Fig. 1.40 Distribution among primary circuit components at the time of vessel failure for PWR station blackout scenario: results obtained running ECART without and with resuspension

STORM facility

The STORM facility is composed by several big components, as shown in the schematic diagram in *Fig. 1.38*. The facility has been devised according to a fully modular design, in order to allow the insertion of a variety of test sections (pipeworks, valves, components, etc.). At present, the STORM test section is dimensioned to allow the study of resuspension in straight pipes, with a maximum velocity of 50 m/s in the widest pipe (203 mm in dia.) and up to sonic level in the narrowest pipe (63 mm in dia.). The most characteristic operating conditions for the STORM plant are given in ***Table 1.2***.

Table 1.2 STORM characteristic operating conditions within the test section

Carrier gas flow rate: – nitrogen – steam	 up to 0.5 kg/s up to 0.5 kg/s
Carrier gas velocity	up to sonic
Gas pressure	up to 0.4 MPa
Gas/wall temperature	400°C
Soluble aerosol mass flow rate	up to 0.5 g/s
Insoluble aerosol mass flow rate	up to 0.5 g/s per each plasma torch
Suspended aerosol mass concentration during the deposition phase	up to 10 g/m^3
Expected aerosol aerodynamic mass median diameter	1.0 - 2.5 μm
Expected aerosol geometric standard deviation	1.9 - 2.3
Surface deposit mass loading	> 20 mg/cm^2

The main components of the plant are:

- Aerosol Generation System, which can produce two types of aerosols, insoluble and soluble. Both the soluble and insoluble simulants will be vaporized using plasma torches.
- Mixing Vessel, with a volume of 10m^3, for the soluble/insoluble aerosol-steam reaction, aerosol formation and aging.
- Steam and Nitrogen Supply System, with a max. rate of 0.5 kg/s for each carrier gas, at 400° C max. temperature and 4 bar pressure.
- The test section system, 5 m long, able to contain 0.063 or 0.208 m diameter pipes, thermically insulated and heated up to 400° C, with the possibility to be cooled down with air at any moment of the test.
- The filtering system, with a quench tank, a cooler and a jet scrubber for the liquid and gaseous waste treatment.

Large-scale integral facilities such as STORM allow experiments to be performed under conditions that closely correspond to real situations. However, due to the large number of variables, such experiments are of complex and difficult interpretation. Difficult but crucial is to make measurements of the physico-chemical properties of the deposited and resuspended debris. It is therefore of prime importance to use extensive and well advanced instrumentation systems. The instruments should measure the required parameter(s) as accurately and precisely as possible without the introduction of additional side-effects. At the same time, any invasive measurement techniques should perturb the aerosol flow as little as possible.

The airborne aerosol particles will be characterised at the entry of the test pipe during the deposition phase and at the exit of the test pipe during the resuspension phase. The characterisation includes the measurement of the aerosol concentration using a total mass filter, the mass based particle size distribution using multiple stage impactors, an advanced laser system for the measurement of particle concentration, particle size distribution and the particle velocity. In addition, a laser sheet visualization system will give more information about the shape of the particles. The advantage of optical systems is that they measure on-line and in real-time.

The characterisation of the deposit during the deposition/resuspension process will be done by using X-ray. The intensity attenuation of a transmitted X-ray beam determines the thickness of the deposit in each measurement position. The uniformity of the deposit in radial and axial direction is obtained by displacing the densitometer. A temporal resolution of 1 s gives quasi a real time measurement. The deposit composition can be determined using X-ray fluorescence.

A series of post-test microscopic and chemical analyses of the deposits in the filters, impactors and on additional deposition coupons is foreseen to complete the data interpretation. This will give information about the deposit composition, the aerosol particles size etc. The thermohydraulic test conditions will be documented using conventional instrumentation for pressure, temperature, flow rate etc.

The Experimental Programme

The facility construction takes place during 1992/1993; concurrently, all required instrumentation will be procured and tested. It is anticipated that the first scoping tests could be carried out between the end of 1993 and the first part of 1994.

The number and priority of the tests to be performed will be established during 1993 in the frame of an international cooperation as wide as possible. The programme will be based on an extensive analytical and experimental preparatory study concerning the phenomenology associated with the injection/deposition processes. As it can be seen from *Table 1.3*, the preliminary test matrix for the scoping tests is mainly focused on the effect of the carrier flow rate and the aerosol soluble fraction on the deposition mechanism. This matrix includes also three tests investigating the resuspension phase.

The two promoting organisations, JRC and ENEL, intend to offer this project for international collaboration inside and outside the EC.

References

[1] MAILLIAT A., SERRE F., JONES A.V., SHEPHERD I. - The Calculation Programme to Prepare the First Phebus-FP Test - USNRC 20th Water Reactor Safety Information Meeting., Bethesda (MD, USA), October 1992

Table 1.3 Preliminary test matrix of the STORM scoping tests, with the test pipe of 63 mm in dia, 5 m long

Scoping Test	Soluble Content (%)	Gas Velocity (m/s)	Resuspension Phase
ST01	-	10	NO
ST02	-	25	NO
ST03	-	50	NO
ST04	-	100	NO
ST05	-	200	NO
ST06	20	10	NO
ST07	20	25	NO
ST08	20	50	NO
ST09	20	100	NO
ST10	20	200	NO
ST11	50	10	NO
ST12	50	25	NO
ST13	50	50	NO
ST14	50	100	NO
ST15	50	200	NO
ST16	-	(*)	YES
ST17	20	(*)	YES
ST18	50	(*)	YES

(*) To be established on the basis of the previous test results

[2] HERRANZ L., VINCENTE M.C., JONES A.V., SHEPHERD I. Contributions to the Pretest Analysis of the Phebus-FP Tests - XVIII Reunion Annual, Spanish Nuclear Society, Jerez, October 1992

[3] SHEPHERD I.M., SERRE F. - Precalculations for the Bundle for the First Phebus-FP test FPT-0 - IAEA Technical Committee Meeting - Behaviour of Core Materials and Fission Product Release in Accident Conditions in Light Water Reactors' - Aix-en-Provence, 16-19 March 1992

[4] CAPITAO J.A., DROSSINOS Y., TAUTGES T., JONES A.V. A comparison of the RAFT and VICTORIA computer codes for simple geometries in chemically reactive systems - J. Aerosol Sci. v22 suppl. 1, pp. 5739-5742, (1991)

[5] SHEPHERD I., JONES A.V.,. HERRANZ L.E, ALCAMI M. - Scoping calculations for the containment in Phebus Test FPT-1 - XVII Reunion Anual de la Sociedad Nuclear Espanõla, Jerez de la Frontera - Puerta de Santa Maria, Spain 28-30 Octubre 1992

[6] SHEPHERD I. et al. - PHEBUS-FPT-0: Calculations for the Reference Scenario, Volume 1: The Bundle EUR 14225EN (1992)

[7] VENTURA M., SHEPHERD I., SERRE F. - Sensitivity calculations for Phebus-FP tests using the ICARE-2 code - Technical note in preparation

[8] FERMANDJIAN J., AKAGANE K., CARLUCCI L.N., CAPITAO J.A., DROSSINOS Y., GONZALEZ C., KISSANE M.P. - Presentation of the Results of Exploratory Circuit Calculations for Phebus FPT-0 - (Reference Scenario), Technical Report SAWG92-035/2 (internal communication), June 1992

[9] CAPITAO J.A., DROSSINOS Y. - Aerosol Interactions and Transport During the First Phebus Experiment According to the VICTORIA and RAFT Computer Codes, J. Aerosol Sci., Vol. 23, Suppl. 1, pp S839-S842, 1992

[10] DUMAZ P., DROSSINOS Y., CAPITAO J.A., DROSIK I. - Fission Product Deposition and Revaporization Phenomena in Scenario with Large Temperature Differences - ANS Heat Transfer Conference, August 1993 (submitted)

[11] JONES A.V., BONANNI E., MARKOVINA A. - Principal results of the Phase B verification studies in support of the Phebus-FP project, CSNI workshop on aerosol behaviour and thermal-hydraulics in the containment - Fontenay-aux-Roses (F), 26-2, 8th November 1990

[12] SHEPHERD I.M. et al. - PHEBUS-FPT-0 Benchmark Calculations - EUR 13698 EN (1991)

[13] ARNAUD A., MAILLIAT A., MARKOVINA A., JONES A.V. - Test matrix of the Phebus-FP programme - revision 2 (internal communication), March 1991

[14] LAYLY V.D., TIRINI S. - FPT-0 Test Protocol Calculations, Containment Thermalhydraulic with JERICHO - Aerosol Behaviour with AEROSOL-B2 (internal communication), Sept. 1992

[15] SHEPHERD I., HERRANZ L., ALCAMI M., SMITH P., ELLICOTT P. - Scoping Calculations for The Thermalhydraulic Control of the Phebus-FP Containment Vessel During High Humidity Transients - EUR report (1993)

[16] JONES A.V., ARNAUD A. - Phebus-FP outline reference geometry and scenario for exploratory calculations: test FPT-2, CEC Joint Research Centre/CEA Cadarache, SAWG92-081/0 (internal communication), September 1992

[17] ALONSO A., GONZALEZ C. - Analysis of tellurium behaviour in the Marviken large scale aerosol transport test with an improved RAFT code version - Polytechnic University of Madrid, EUR 13792 EN, 1992

[18] WILLIAMS D.A. - Summary report of Phase II of the VICTORIA code development programme - AEA Technology Winfrith, AEA RS 5329, April 1992

[19] HEAMES J.J. et al. - VICTORIA: A Mechanistic model of radionuclide behaviour in the reactor coolant system under severe accident conditions, NUREG/CR-5545, Rev. 1 (Dec. 1992)

[20] JONES A.V., SHEPHERD I. - ESTER - a Eurpean Severe Accident Code System - USNRC 20th Water Reactor Safety Information Meeting, Bethesda (MD, USA), October 1992

[21] HOLDFORD G.F., WHEATLEY C.J. - VICTORIA aerosol-atmosphere interaction model development - AEA Reactor Services Risley, AEA RS 5226, February 1992

[22] HONTANON E. - Aerosol nucleation in severe nuclear accidents: scenarios of interest and modelling - Polytechnic University of Madrid, CTN-56/92, 1992

[23] DEANE A.M. et al. - Development of Models to Follow Vapour-Aerosol Rections and Iodine Chemistry - AEA RS 5327, July 1992

[24] BELL J.T. et al. - Predicted rates of formation of iodine hydrolysis species at pH levels, concentrations, and temperatures anticipated in LWR accidents - NUREG/CR-2900, ORNL-5876 (Oct. 1982)

[25] TOTH L.M. et al. - The chemical behaviour of iodine in aqueous solutions up to 150°C - NUREG/CR-3514, ORNL/TM-8664 (April 1984)

[26] FERMANDJIAN J. - FPT-0 test, Iodine chemistry in the containment vessel. Value of sump water pH - Fiche technique PHEBUS-FP, PFR 040C/CPF 097C (internal communication), October 1992

[27] M. EUSEBI, F. PAROZZI, M. VALISI, J.A. CAPITAO, G.F. DE SANTI - Preparatory Calculations for a New Experimental Program on Dry Aerosol Resuspension Mechanisms (STORM Project) - Paper presented at the European Aerosol Conference, Oxford (UK), September 1992

[28] G. AGRATI, et al. - STORM: Simplified Tests on Resuspension Mechanisms - Preliminary Description of the Project, ENEL/DRS/CRT Report N6/91/01/MI, 1991

[29] G.A. SEHMEL - Particle Resuspension: A Review Environmental Intern., Vol 4, 1980

[30] H.A. MOREWITZ, et al. - Resuspension/Re-entrainment of Aerosol in LWRs Following a Meltdown Accident, GREST Draft Report, 1986

[31] F.J. RAHN, J. COLLEN, A.L. WRIGHT - Aerosol Behaviour Experiments on Light Water Reactor Primary Systems, Nuclear Technology, Vol. 81, no. 2, 1988

[32] A. FROMENTIN - Particle Resuspension from a Multi-Layer Deposit by Turbulent Flow, Paul Scherrer Institut, Report no. 38, Würenlingen (published also as LACE Report TR-083), 1989

[33] R. BOLADO, E. HONTANON - Aerosol Resuspension in the Reactor Cooling System of LWR's Under Severe Accident Conditions. State-of-the-Art Report, Universidad Politecnica de Madrid, EUR 13789 EN, 1991

[34] F. PAROZZI - Development of Computer Models on the Chemical and Physical Behaviour of Fission Products and Aerosols in the Primary Coolant System of a LWR Plant During a Severe Accident. Part II, FP and Aerosol Transport, EUR 14676/II EN, 1992

1.1.2 THE FARO LWR PROGRAMME

The FARO plant

The JRC-Ispra FARO plant is a large multi-purpose test facility in which phenomena related to severe accidents in Nuclear Reactors can be simulated. Basically, a maximum quantity of the order of 150 kg of oxide fuel type melts (up to 3000°C) can be produced in the FARO furnace, possibly mixed with metallic components, and delivered to a test section of interest (containers up to 1.0 m^3, 10 MPa and 300°C are available). From 1987 to 1990, the plant has been used for LMFBR safety problems such as melt relocation and molten fuel/sodium interaction.

Following the recommendations of the EC-Member States experts, a change to LWR Severe Accident problems was decided in 1990. A first series of corium melt/water quenching experiments was launched in September 1990 in collaboration with the United States Nuclear Regulatory Commission (US-NRC) and the Electric Power Research Institute (EPRI). This test series is part of the 1992-94 CEC Framework programme and is discussed and assessed with a group of EC experts.

The reference situation is a postulated core melt down accident when jets of molten corium penetrate into the lower plenum water pool, fragment and settle on the lower head. There is a lack of data on the issue, and particularly on the water quenching potential that determines whether the thermal loading on the bottom head structures can be mitigated. It has been agreed to perform experiments for investigating this melt/water mixing and quenching process and melt/structure interaction for different pressures, numbers of jets and bottom vessel geometries. The FARO facility, which has the following unique test features:

- large masses of molten corium (up to 150 kg) of prototypical compositions ($UO_2/ZrO_2/Zr$),
- full-scale water depth (up to 2 m),
- high pressure, high temperature interaction vessel (TERMOS: 10 MPa, 300°C), has been considered suitable for such investigations.

The objective of the present test series is to determine, for different pressures, bottom vessel geometries and number of jects:

- the steam generation rate associated with the melt quenching,
- the hydrogen production associated to the zirconium oxidation,
- the thermal load on the bottom structures. The first part of the test matrix deals with high pressure scenarios. A first quenching test (Scoping Test) with 18 kg of oxide melt quenched into 1 m depth water was performed in 1991 [1,2].

During 1992, the following main activities have been performed:

- Execution of a test to check the capability to produce, deliver and measure the temperature of 150-kg UO_2-ZrO_2-Zr melt ((Jet Characterization Test),
- Execution of a second quenching test with 44 kg of UO_2-ZrO_2 melt (Quenching Test 2),
- Decontamination of the test room after the TERMOS blow-down that followed the Quenching Test 2 (QTZ),
- Completion of the analysis of the Scoping Test and issue of the Quick Look and Data Reports,
- Preparation of the facility for the 150 kg melt quenching tests:
    - Start of installation of the steam venting unit (steam-water separator, venting piping, condenser),
    - Study, design and start of manufacture of new components for the test vessel: release vessel, debris catcher, instrumentation rack, heating unit.

Jet Characterisation Test

The test was performed on 2nd April. It was a simulation, in dry conditions, of the melt path expected during the first 150 kg melt quenching test (Base Case Test). The objectives were to produce 150 kg of 76w% UO_2 - 19w% ZrO_2 - 5w% Zr molten corium, to measure the temperature of the melt, and to visualise the jet during delivery in a steam dense atmosphere simulated by SF_6 at 0.4 MPa. The two first objectives were met, but the jet visualisation failed due to SF_6 thermal decomposition.

The test arrangement is shown in *Fig. 1.41*. The release vessel (S/S) was connected to the UO_2-ZrO_2 melting furnace (FARO furnace) via the release channel and the isolation valve unit (SO1, SO2). The release vessel contained about 1000 m of 1.2 mm diameter Zr wire (7 kg) uniformly distributed in the whole volume. The superheated oxide melt coming from the furnace induced the melting of the zirconium and the formation of a homogeneous mixture. After closing the SO2 valve and pressurising of the test region to 0.4 MPa using SF6, the corium was delivered to the test vessel through the special slide valve SO4 for visualisation. It was then collected in the collector vessel (S/S). Two series of two vertically mounted tungsten ultrasonic sensors (UTS) measured the melt temperature in the release vessel and in the collector vessel, respectively.

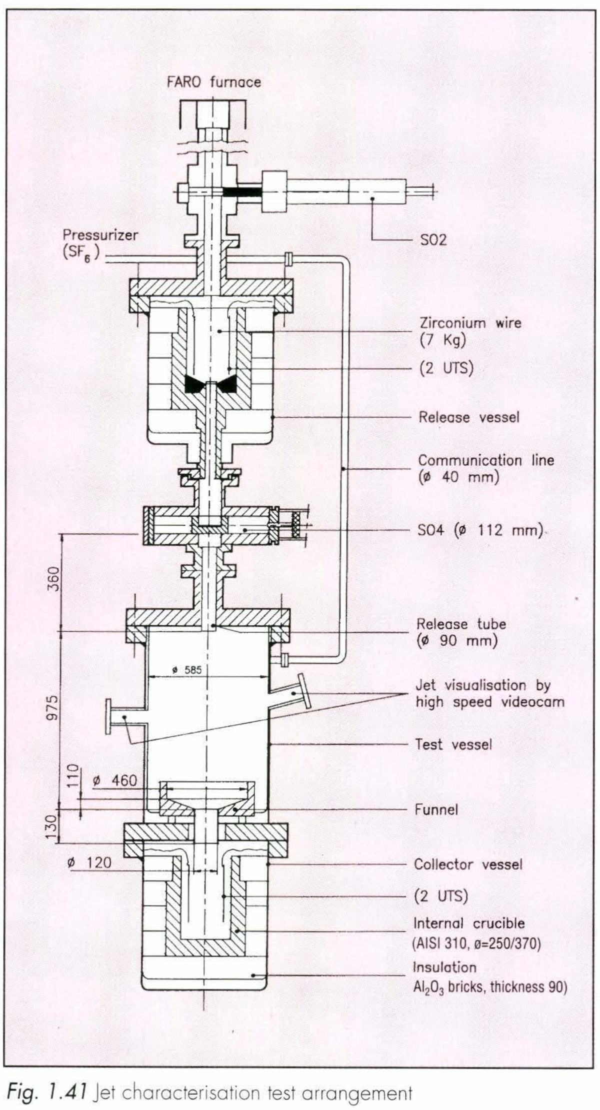

Fig. 1.41 Jet characterisation test arrangement

The zirconium melting procedure worked as expected. Approximately 135 kg of corium were produced in the release vessel, from which 112 kg arrived in the collector vessel. The rest formed crusts in the release vessel and in the test vessel funnel. *Fig. 1.42* shows the corium temperature as measured by the UTSs. A maximum temperature of 2500°C was recorded. After approximately 1 minute the UTS failed, probably as a result of chemical incompatibility between tungsten and zirconium at such high temperatures. Nevertheless, the test demonstrated the feasibility of the measurements.

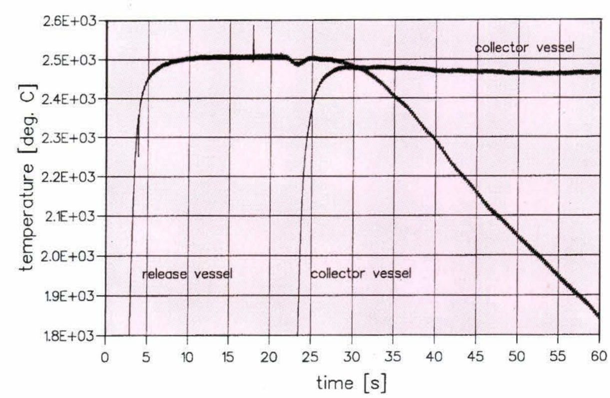

Fig. 1.42 UTS traces: jet characterisation test

Post-test examination showed that the UTS portions that were immersed in the corium had completely disappeared. No traces of erosion of the release vessel were observed. The collector vessel, in which the melt completely solidified, is being analysed (cuts are being made for characterising the debris and structures). A corium crust some millimeters thick covering almost the entire funnel (larger diameter 460 mm) was found, indicating that significant melt break-up and spreading occurred. The distance between the SO4 outlet tube and the funnel being comparable to the melt through steam free fall in the Base Case Test (~ 1 m), a similar spreading before the melt contacts the water is to be expected in this test.

Second Quenching Test (QT2)

The test was performed on 29th July. The objective was to obtain further information on the melt/water quenching process by making a further step in the melt quantity with respect to the Scoping Test (ST). 44 kg of 80w% UO_2- 20w% ZrO_2 melt (against 18 kg for ST) were released into 255 kg of slightly subcooled water at 5.8 MPa (against 120 kg for ST) contained in the TERMOS interaction vessel of the diameter 0.71 m (against 0.47 m for ST).

The test arrangement is shown in ***Fig. 1.43***. The interaction vessel TERMOS was connected to the UO_2-ZrO_2 melting furnace via the release channel and isolated from it during interaction by the intersection valve SO2. The melt was first delivered to the release vessel, and then released to the water. Initially, the release vessel was at the same low pressure as the furnace (0.2 MPa). After transfer of the UO_2-ZrO_2 mixture to the release vessel, the intersection valve SO2 was closed and the release vessel pressure balanced to the TERMOS pressure using the communication line. To suppress boiling of the water during the pressure equalization, argon was blown into TERMOS just before bursting the communication line rupture disc. Upon pressure balancing, the melt catcher tap automatically opened. By this procedure, the melt was released by gravity to the water and collected in the debris catcher.

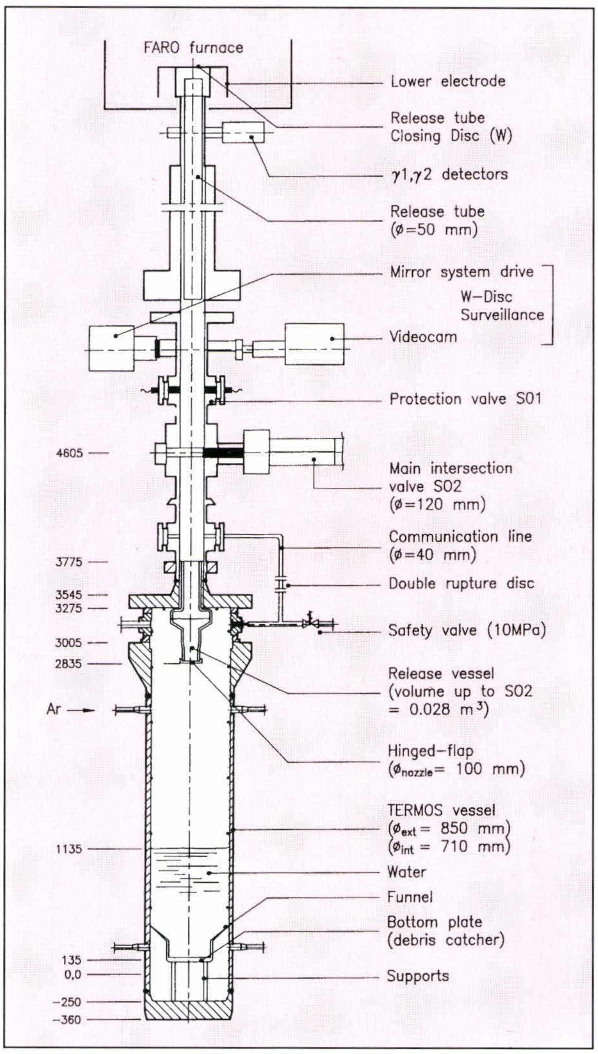

Fig. 1.43 Quenching test 2 arrangement

The principal quantities measured during the corium quenching were pressures and temperatures both in the gas region and in the water, and temperatures in the debris catcher bottom plate. The distribution and types of the probes are reported in ***Figs. 1.44*** and ***1.45*** for the test vessel and the debris catcher, respectively. Four KELLER pressure transducers (piezo-resistive, 5-kHz frequency response) measured the vessel pressurization. Four VIBROMETER's (piezo-electric, 15-kHz frequency response) were located in the water as indicated in ***Fig. 1.44*** (radial position: 195 mm from the vessel centre-line) for rapid transient records in case of an FCI. They were protected by stainless steel grids. The eight steam K-thermocouples were installed in such a way that they could not "see" a centred melt jet. 14 thermocouples were fixed on the structures by means of clamps. The 25 water K-thermocouples were essentially sacrificial thermocouples used to determine the downward progression and radial extension of the melt jet. Those not destroyed during melt penetration recorded the long time water temperature history. They were attached on thin (0.2 mm) stainless steel wires crossing the test section. The opening of the

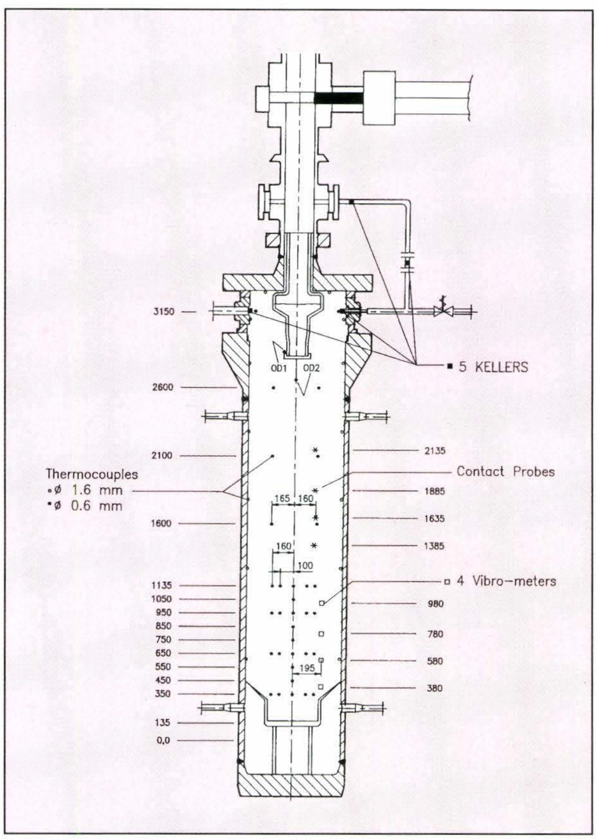

Fig. 1.44 Quenching test 2 instrumentation

melt catcher was indicated by the rupture of a 0.5 mm K-thermocouple (OD1) mounted opposite to the hinge, and fixed both to the melt catcher lower flange and to the flap. Another 0.5 mm K-thermocouple (OD2) was placed on the centre-line of the vessel, 250 mm below the lower face of the flap, for detecting the passage of the melt. Four resistance probes for measuring the level swell were installed 250, 500, 750 and 1000 mm above the initial water level (radial position: 160 mm from the vessel centreline). The experimental conditions are summarised in ***Table 1.4*** and compared to the scoping test conditions.

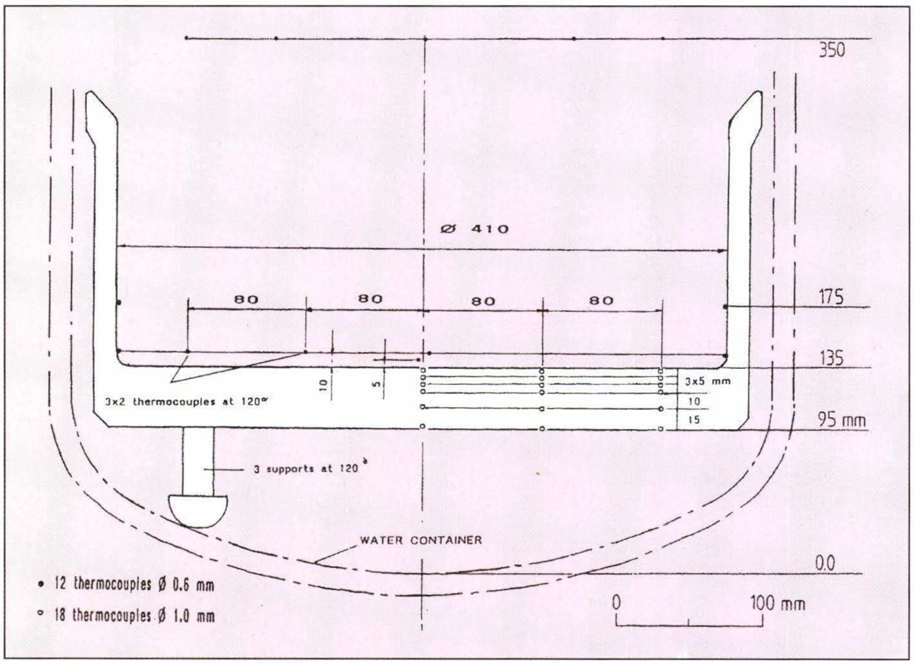

Fig. 1.45 Debris catcher instrumentation

The results are summarised in ***Table 1.5*** and compared to the Scoping Test (ST) data (for details see [3]). For all the data reported, time zero corresponds to the OD1 rupture (melt catcher hinged-flap opening, i.e. start of melt release). The cover gas pressure traces corresponding to the first 1.4 seconds of the test (roughly corresponding to the melt fall and break-up stage) are reported in ***Fig. 1.46*** and compared to that of the scoping test. The values are normalised to the value at time zero, which were 5.0 MPa for ST and 5.8 MPa for QT2. The curves indicate that the break-up phase was essentially the same for both tests. The pressure increase after melt/water contact (MWC, deduced from thermocouples) was 1.5 MPa against 0.7 MPa for ST. The end of break-up time (EOB) reported in the figures means that the unbroken upper part of the trailing edge should have reached the bottom plate at that time. It has been calculated by adding the delivery time to the time at which the leading edge touched the bottom (MBC). The water pressure transducers gave exactly the same signals as the cover gas transducers, except the classical late drift due to heating which indicates that no steam explosions occurred.

Fig. 1.47 shows the signal of the resistance probe located 250 mm above the initial water level for QT2. Only this probe signal moved from the rest, which gives an indication of the maximum level swell.

The early steam temperature increases are presented in ***Fig. 1.48***. As can be seen, the steam temperature was rather non-uniform, which suggests a non-uniform steam generation and mixing with the cover gas during the break-up phase. Estimates of the amount

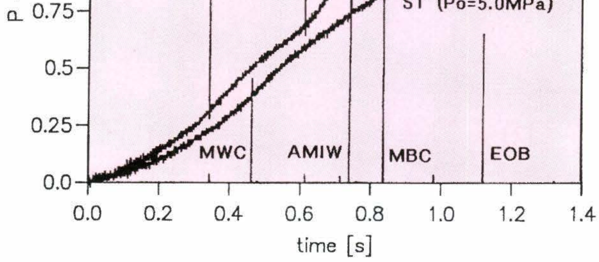

Fig. 1.46 Early vessel pressure increase (ST: Scoping Test, QT2: Quenching Test 2, t=0: release start, MWC: Melt/Water Contact, MBC: Melt/Bottom Contact, AMIW: All Melt in Water, EOB: End of Break-up

Table 1.4 Summary of QT2 experimental conditions compared to ST

	Scoping Test	**Quenching Test 2**
Melt		
Composition, w%	80 UO_2+20 ZrO_2	idem
Mass, kg	18	44
Temperature, °C	2650	2750
Delivery nozzle, mm	100	100
Delivery time, s	0.280	0.370
Flow rate, kg/s	64	119
Free fall in gas, m	1.83	1.70
Water		
Mass, kg	120	255
Depth, m	0.87	1.00
Temperature, °C	266 (230 bottom plate)	263 (255 bottom plate)
Subcooling at melt contact, °C	2 (38 bottom plate)	12 (20 bottom plate)
Fuel to coolant mass ratio	0.15	0.17
Gas Phase		
Composition, w%	83 steam + 17 Ar	70 steam + 30 Ar
Volume, m^3	0.464	0.875
Temperature at OD1 rupture (release start), °C	270	263
Test Vessel		
Diameter, m	0.470	0.710
Overall volume, m^3	0.640	1.3
Pressure at OD1 rupture, MPa	5.0	5.8
Pressure at melt/water contact, MPa	5.4	6.1

of steam produced during the break-up were made by using the steam table at the partial pressures of the steam corresponding to the pressure increases after melt/water contact (i.e. between 4.8 and 6.1 MPa for ST, and between 4.9 and 6.4 MPa for QT2). Values of 1.7 kg for ST and 7.4 kg for QT2 at Tg=300°C were found, corresponding to a transferred energy of 2.5 MJ for ST and 10MJ for QT2.

The pressure history over a period of 25 seconds is reported in *Fig. 1.49* and compared to that of the scoping test. The long term maximum pressure was reached after approximately 22s, against 12s for ST. The first maxima separate the two characteristic phases of the tests, namely the short-term break-up phase and the long-term debris cooling phase. In *Fig. 1.50* the cover gas and water temperature histories for 25 s are plotted. Assuming that the 20°C water temperature early jump observed in *Fig. 1.50* is the result of the melt quenching during the fall stage, an energy of 25 MJ transferred to the water during that phase is calculated. By adding these 25 MJ to the estimate of the energy released to the steam (10 MJ), approximately 50% of energy transferred to the steam/water system during the melt fall stage are obtained, which corresponds globally to smaller scale similar tests (CCM series [4]).

Table 1.5 Summary of QT2 experimental results compared to ST (time t=0=release start)

	Scoping Test	**Quenching Test 2**
Melt		
Mean velocity in cover gas, m/s	4	5
Mean velocity in water, m/s	2.3	3.7
Fragmented, kg	12	30
Molten on bottom plate, kg	6	14
Mean size of fragments, mm	4.5	3.8
Melt/debris fluidization	non	non
Delivery nozzle, mm	100	100
Bottom Plate		
Maximum temperature increase, °C	–	275 (contact face)
Maximum downward heat flux, MW/m^2	–	0.8
State	intact	intact
Pressure Increase		
Maximum short-term from release start, MPa	1.1 (at t=1.2s)	1.8 (at t=1.2s)
Maximum short-term from melt/water contact, MPa	0.7	1.5
Maximum long-term contact, MPa	1.6 (at t=12s)	1.8 (at t=22 s)
Steam explosion	non	non
Temperature Increase		
Steam (maximum measured), °C	86	83
Steam (mean value at t=10 s), °C	~ 43	~30
Water	15 (maxi at t=12 s)	23 (at t=25 s)*
Level Swell		
Level swell, mm	130**	250 (~ maxi)

* maximum not reached at that time - data not available beyond
** from thermocouples - not necessarily the maximum

Beyond 25s, a rupture of the by-pass line caused by a melt deposit near to the connection with the release vessel led to a blow-down of the pressure vessel and to the contamination of the test room. The debris structure looked very similar to that of the scoping test and seemed not to have been influenced by the vessel blow-down. About 2/3 of the melt broke up in particles of a mean size of 3.8 mm (*Fig. 1.51*) according to the particle size distribution of *Fig. 1.52* (contrarily to pure UO_2, UO_2-ZrO_2 fragments are very brittle and great care must be taken in manipulating them in order not to alter the analysis). The broken up part was located above a conglomerate corresponding to about 1/3 of the melt (visible on top of *Fig. 1.53*). The conglomerate was certainly still molten when contacting the bottom plate, and solidified on it. Nevertheless, the plate did not suffer any damage, as can be seen in *Fig. 1.54*. This is not surprising using a pure oxide melt because it was known already, from the BLOKKER series performed in FARO in the frame of the LMFBR programme [5], that jets of 100 kg of pure UO_2 around 3000°C interacting in dry conditions with 40 mm thick plates preheated to 400°C, did not induce any erosion even though the plates reached 1000°C.

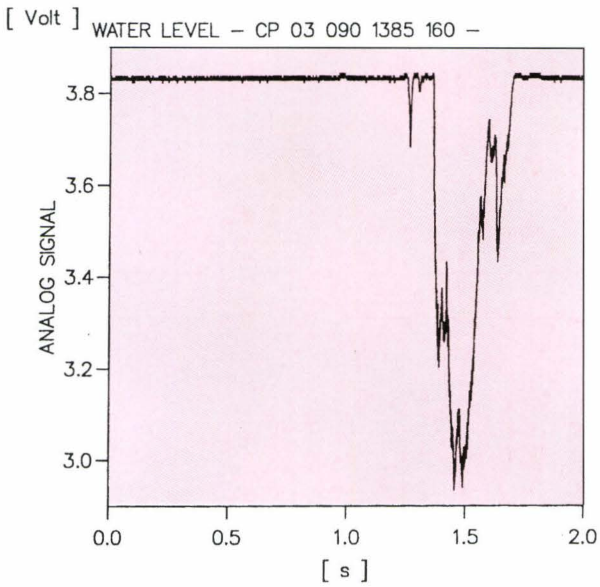

Fig. 1.47 Signal from the QT2 level indicator located 250 mm above the initial water level

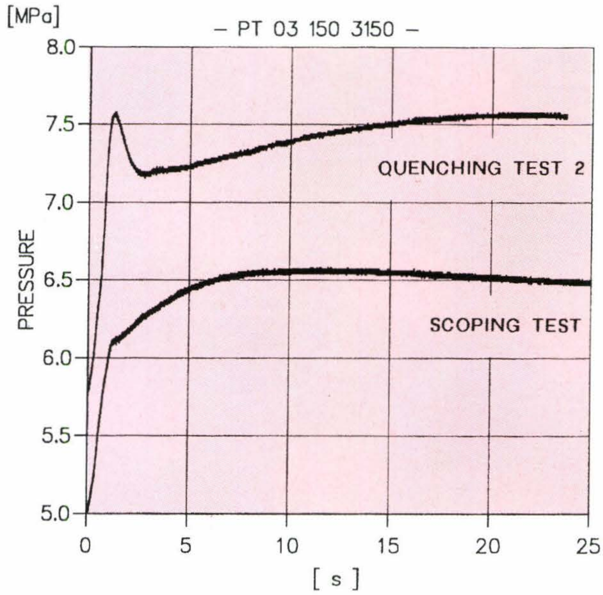

Fig. 1.49 Long-term pressure histories

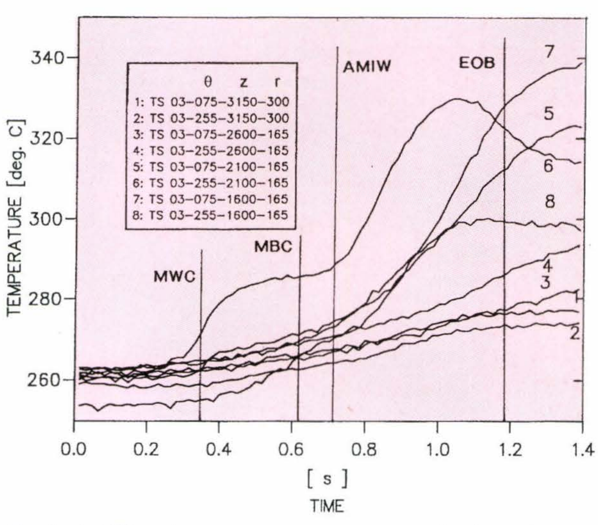

Fig. 1.48 QT2 early steam temperature increase (for legend see Fig. 1.46)

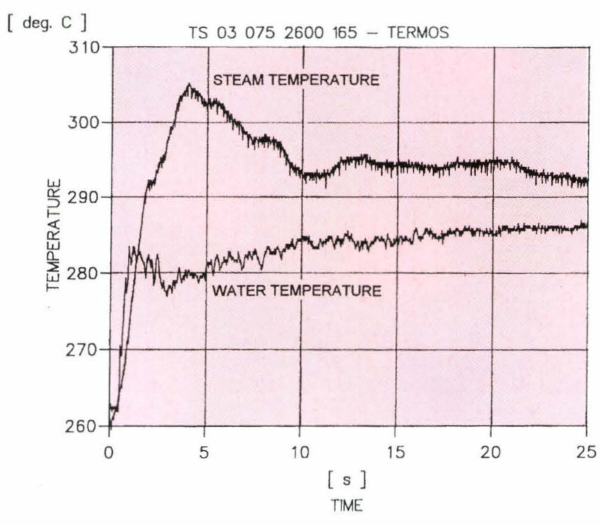

Fig. 1.50 Quenching test 2 steam and water temperatures

Most of the thermocouples located just above the surface of the bottom plate were destroyed during melt impact. Contact temperatures and those 5 mm below the bottom plate surface are shown in ***Fig. 1.55*** for different radial locations. They reflect the fact that the melt did not spread uniformly on the plate. Downward heat fluxes calculated from these temperatures are reported in ***Fig. 1.56***. A maximum value of 0.8 MW/m^2 is obtained. The time integrals of these curves give the energy release to the plate (***Fig. 1.57***), which reached a maximum around 9 MJ/m^2 at time 20s.

In conclusion, the essential objectives of the test were achieved. No fundamental differences were noted with respect to the scoping test (see also Ref. [3] for further details on the comparison ST/QT2). This enhances the need for tests with larger amounts of melt. In addition, the influence of the presence of a percentage of a metallic compound in the melt has still to be evaluated. The first of such a test with 150 kg of prototypical corium (Base Case Test) will be performed under conditions as close as possible to that of the previous tests (closed volume), in order to facilitate comparisons and code validation. The execution of this test will be one of the major tasks for 1993.

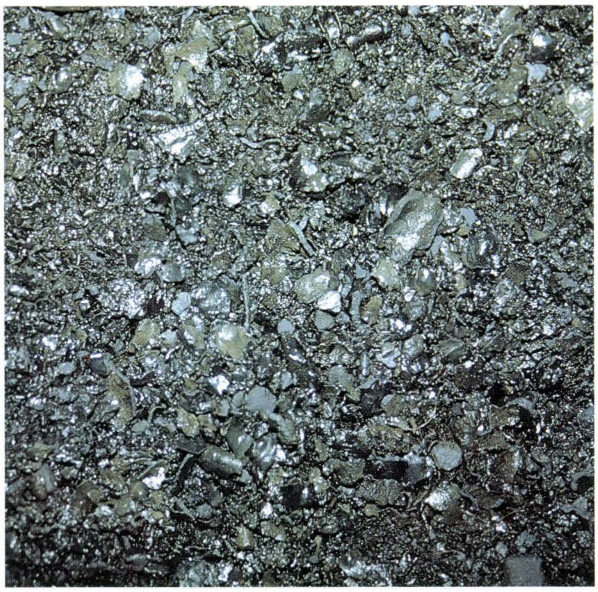

Fig. 1.51 View of the debris bed from above

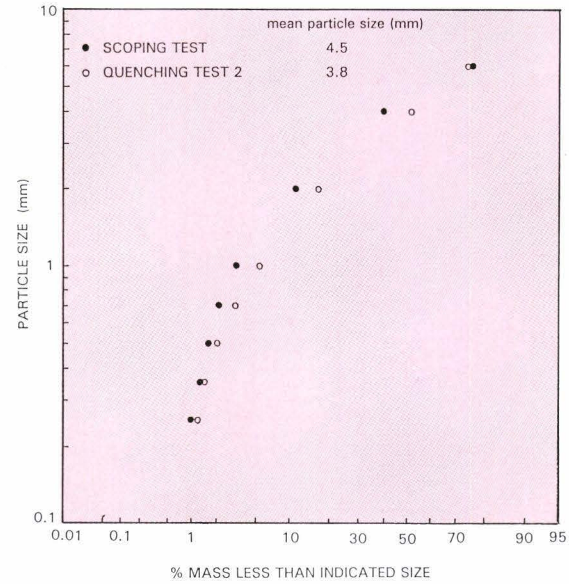

Fig. 1.52 Particle size distributions for scoping test and quenching test 2

Fig. 1.53 View of the debris bed after partial removal from the catcher

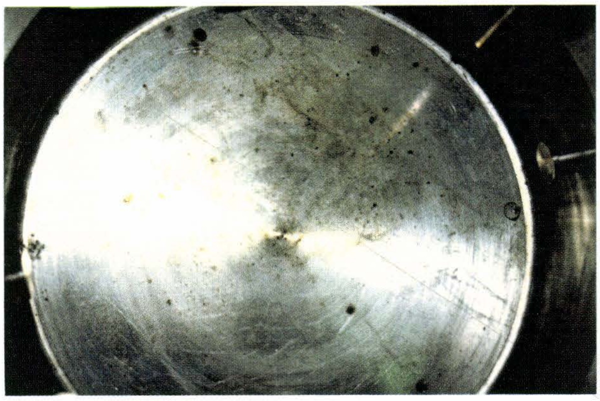

Fig. 1.54 Bottom plate after debris removal

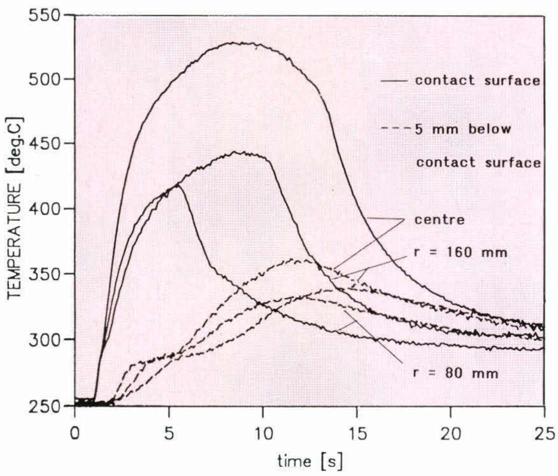

Fig. 1.55 Melt/bottom plate contact temperatures and temperatures in the plate 5 mm below the contact surface

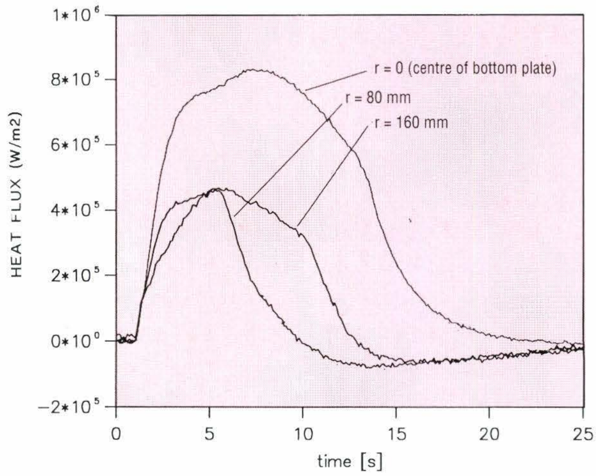

Fig. 1.56 Downward heat fluxes at different radial locations

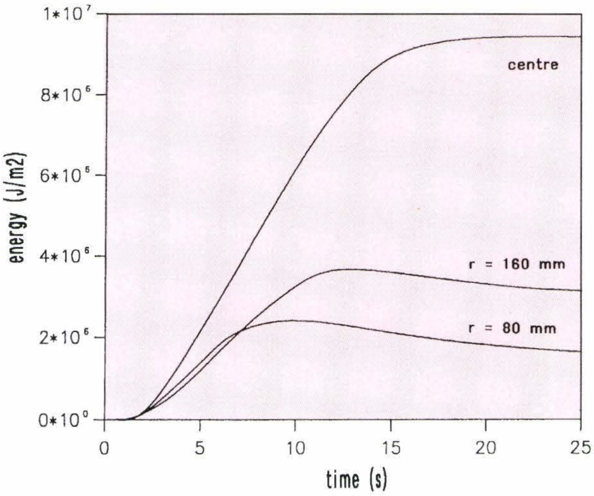

Fig. 1.57 Energy transferred to the bottom plate (quenching test 2)

Preparatory Work for the Base Case Test (BCT)

This test will be performed at 5.0 MPa with 150 kg of prototypical corium poured into 560 kg of water of 2 m depth. The preparatory work concerned mainly the Venting Unit assembly and the TERMOS vessel modifications. In addition to that work, the decontamination of the test room after the Quenching Test 2 blow-down event was initiated soon after the test by Ispra specialists and the contamination was brought down to zero by an external firm (end of September).

The first two quenching tests using relatively small quantities of melt were performed in a totally closed vessel. With 150 kg of melt, a venting of the steam during the quenching will be available. The venting unit (*Fig. 1.58*) will include a steam/water

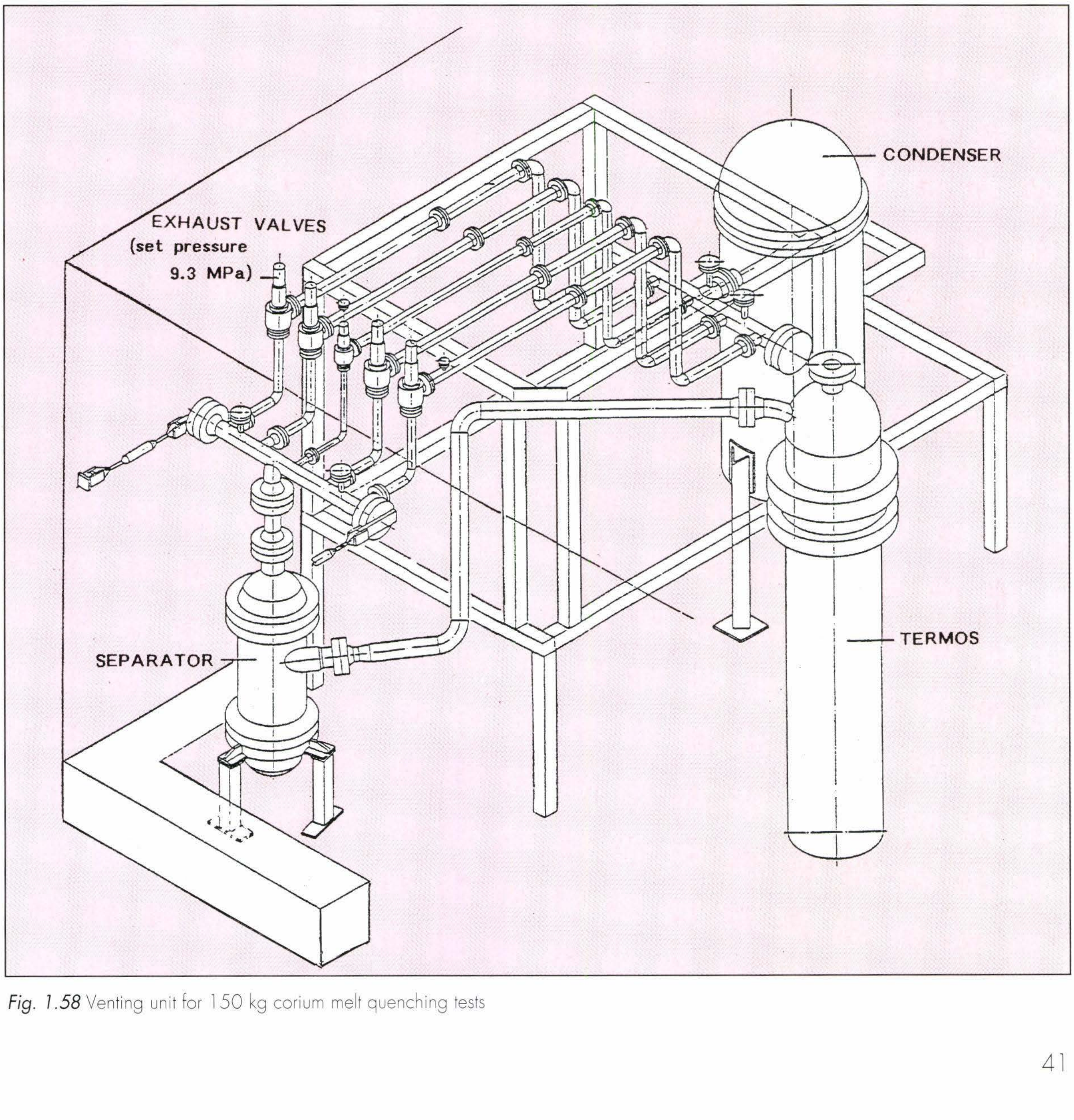

Fig. 1.58 Venting unit for 150 kg corium melt quenching tests

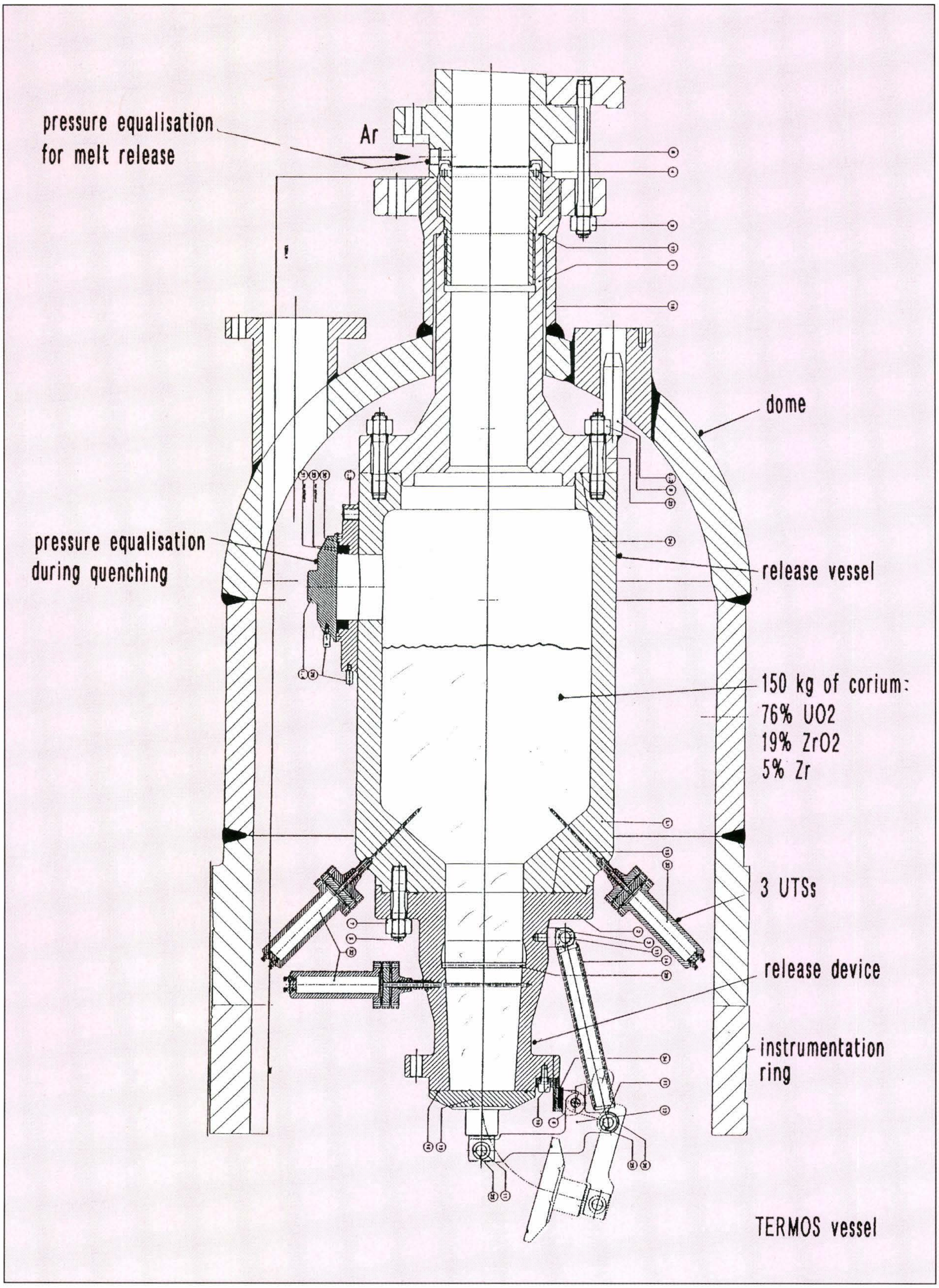

Fig. 1.59 Release vessel for 150 kg corium melt quenching tests

separator, 4 exhaust valves (safety valve type, set pressure 9.3 MPa) and a condenser. It will act as a safety device to prevent a possible beyond design pressurisation of TERMOS (design pressure 10 MPa). The procedure will allow to work as far as possible in a closed volume, and will facilitate steam production evaluation and comparisons with the previous tests.

The design and drawings of the Venting Unit were finalised in the first semester. The installation of the components and piping could start only at the beginning of October after decontamination of the test room. It is almost completed. The next steps are the installation of the heaters and the thermal insulation, which should be completed by the end of February '93. The valves have been returned to the manufacturer for modifications and should be ready for the very beginning of 1993.

The analysis of the blow-down event that followed Quenching Test 2 induced the need to substantially modify the gas path between TERMOS and the release vessel. A new concept that eliminated the external communication line was studied (*Fig. 1.59*). Validation tests were launched. The fabrication of a new release vessel accounting for the modifications and capable to hold 150 kg of melt started in November. As a consequence, the base case test execution originally planned for the end of 1992, will be delayed by 3-4 months.

Texas-II Developments

Comparison between experimental data from the FARO/LWR scoping test [3] and early predictions obtained by the computer code TEXAS-II [6] developed in collaboration with the University of Wisconsin revealed some modelling deficiencies. It was clear from the results that the predicted rate of vapour production at the debris bed was too large. Recent calculations performed, using a simplified agglomeration model for the debris bed, have resulted in better agreement with experiments. A more correct estimation of the initial jet diameter has led to better timing for the fuel particles to fall through the water to the bottom of the vessel.

Further improvements of the graphical tool have led to better understanding of the physical phenomena and faster analysis. *Fig. 1.60* illustrates a typical

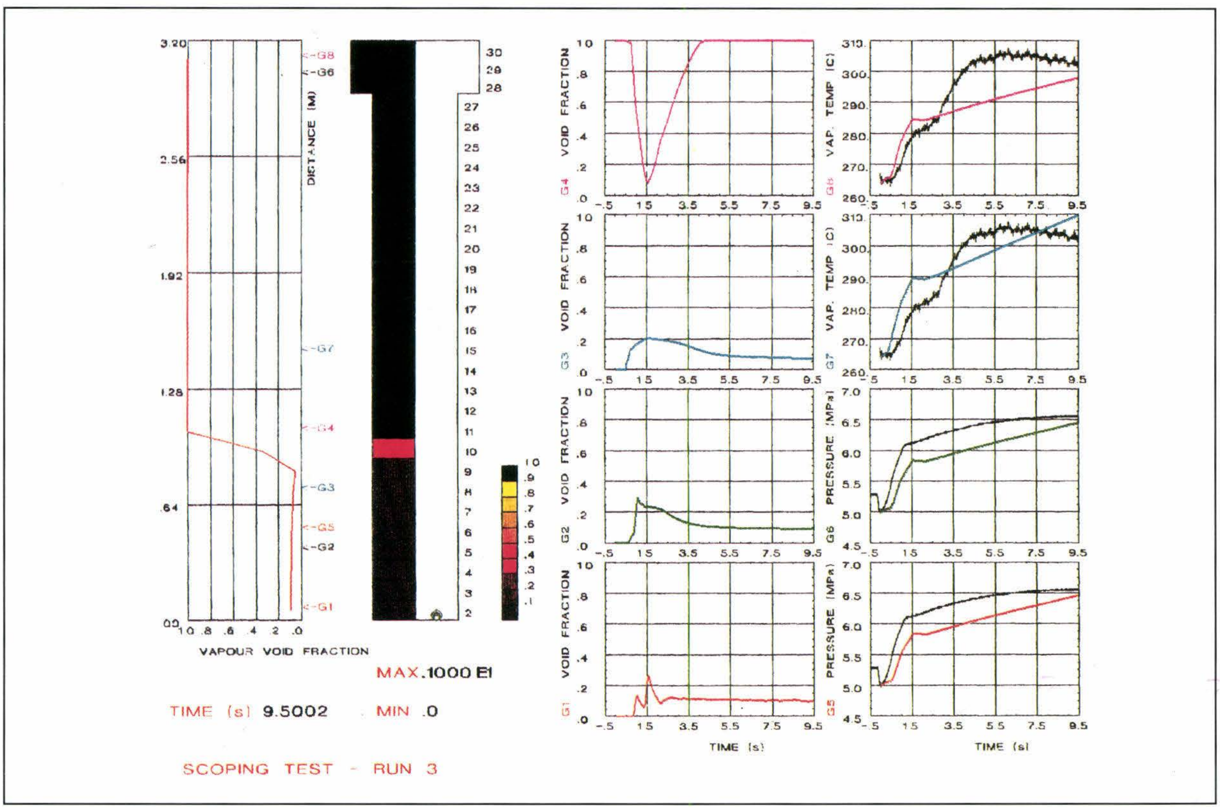

Fig. 1.60 Graphical output for TEXAS analysis - scoping test

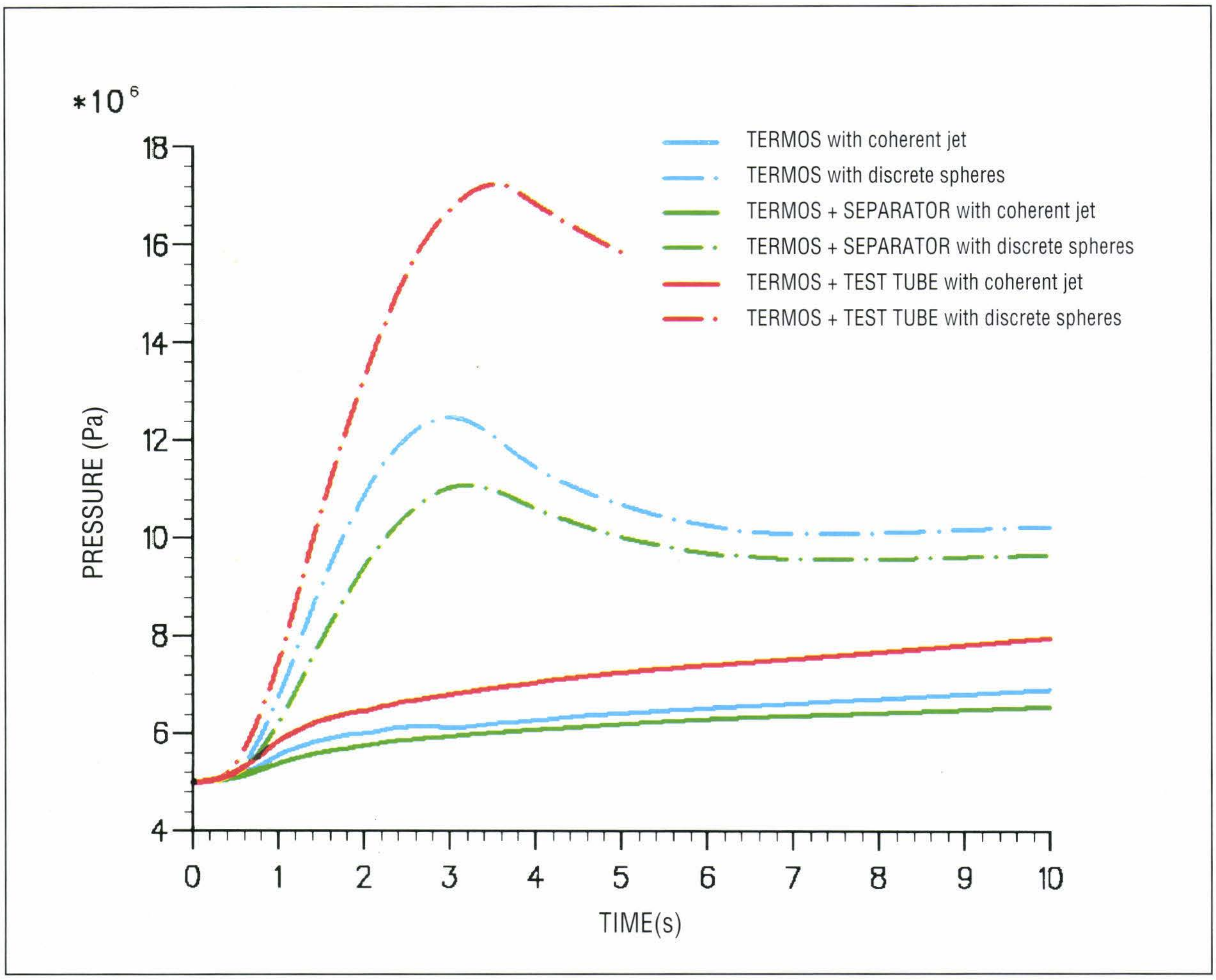

Fig. 1.61 Pressure comparison - base case pre-calculations

composite graph obtained during a scoping test calculation. Eight variable/time graphs (right hand side) show the development of void fraction, pressure and vapour temperature at various positions in the TERMOS vessel and compares them directly with available experimental data. The predicted pressure (graphs G5 and G6) and vapour temperatures (graphs G7 and G8) are in quite good agreement with experiment (black curves). Animated graphics (left hand side) during the transient show how the vapour fraction varies in time and space. Further animated graphics (centre) during the transient show how the fuel (represented as spheres) penetrates the water and forms a debris bed at the bottom of the TERMOS vessel. The degree of colour shading in the Eulerian cells indicates roughly the values of the vapour fraction and any notable variation in water level can be easily seen.

Pre-calculations for the FARO/LWR second quenching test (50 kg fuel at 3000 K, 392 l water at 537 K, 5 MPa initial uniform pressure) predicted results similar to the first scoping test. This was hardly surprising because the fuel/water volume ratio is roughly the same in both cases (scoping test 1:58, 2nd quenching test 1:63) and TEXAS-II is a 1-D code.

Revised pre-calculations for the FARO/LWR Base Case Test (15 kg fuel at 3000 K, 2 m water at 537K, 5 MPa initial pressure) with different TERMOS geometries (with and without separator, with internal test tube) have been performed assuming constant volume conditions. Two molten fuel jet models, coherent jet and discrete spheres, were used. The heat exchange from fuel to coolant is much lower in the former model because only the leading edge of the jet can fragment. The latter model, although not realistic in practice, does provide a useful upper bound estimate for the fuel coolant interaction (FCI). ***Fig. 1.61*** shows the pressure histories over 10s for six calculations. With smaller volume of water and discrete sphere jet the pressure is high, however, it is not intended to install the internal test tube in the TERMOS vessel.

IFCI Developments

The Integrated Fuel-Coolant Interaction Code (IFCI) developed at Sandia National Laboratories (SNL) [7] is now available at JRC. This code, like TEXAS-II, will also be used to investigate FCIs in a mechanistic manner, including coarse fragmentation and mixing of molten fuel with water, triggering, propagation and fine fragmentation, and expansion of the melt-water system.

IFCI is based on MELPROG's two-dimensional, cylindrical geometry, four field hydrodynamic method [8]. Use is made of a multifluid method with separate mass, momentum, and energy equations for each field. The constitutive relations required for the interfield coupling terms (heat transfer, momentum exchange and phase change) include the bulk boiling model, a subcooled surface boiling model, a three-field flow regime map, adaptations of standard heat transfer and momentum transfer correlations.

Major problems were encountered in trying to create a workstation double precision version of the Cray version. To by-pass this obstacle the original version will now be run on a new 64-bit Digital workstation. Further joint research and development of IFCI will be done in collaboration with the University of Wisconsin and ENEL.

RELAP5/MOD3 Calculations

The RELAP5/MOD3 thermalhydraulic computer code [9] has been applied to the FARO facility. Calculations have been performed to understand the quantities measured during the Scoping Test and for the preparation of the Quenching Test [10] and the Base Case Test [11].

The effect of radiation from the falling melt on the experimental thermocouples has been analysed simulating the response of thin thermocouples immersed in the gas phase under the conditions of the FARO Scoping Test. The calculations indicated that a substantial superheating can be measured by the thermocouples with a temperature even much greater than both the steam or the liquid calculated temperature (*Fig. 1.62*). This effect explains the spread of the temperatures really measured during the test.

The second FARO Quenching Test should have been performed changing the initial procedure. The new procedure consisted in pressurizing the TERMOS vessel before realizing the pressure equalization with the melt catcher. In order to verify how great this pressurization should have been, ad-hoc calculations were performed by the RELAP5-MOD3 code. These calculations indicated (*Fig. 1.63*) that a 2.5 bar pressure increase guaranteed enough subcooling margin for the subsequent pressure decay and gave confidence to the experimentalists in the proposed procedure.

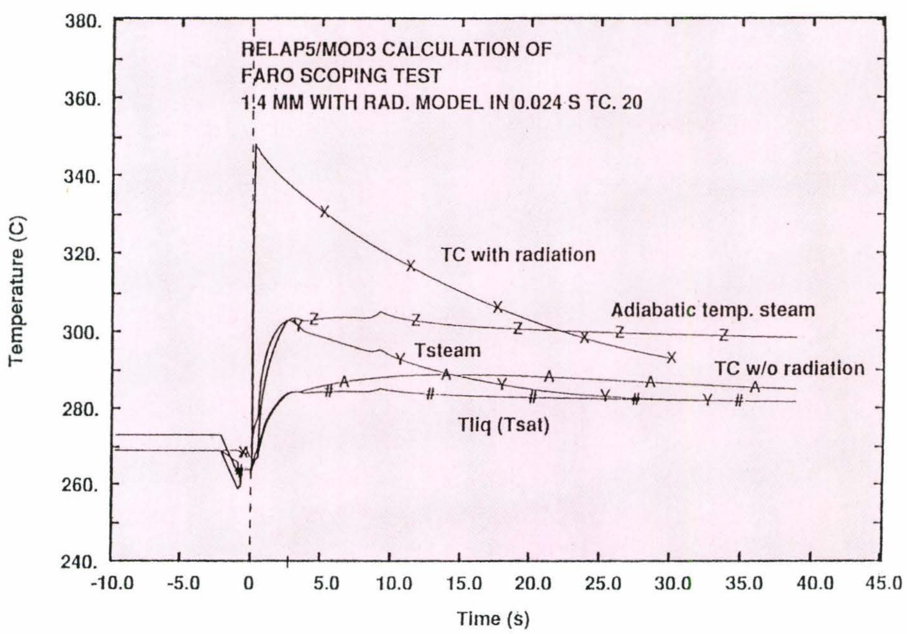

Fig. 1.62 Effect of radiation heat transfer on experimental thermocouples during scoping test 1 of FARO facility

Other pre-test evaluations have been performed in order to verify the dimension of the geometrical configuration of the equalization orifice for the Base Case Test to be performed in 1993. The calculations indicated (*Fig. 1.64*) that the proposed valve diameter was sufficient to allow a stable discharge of the melt catcher volume. At the same time they gave indication that the equalization valve should open earlier than the melt discharge valve otherwise this valve could not open at all during the test.

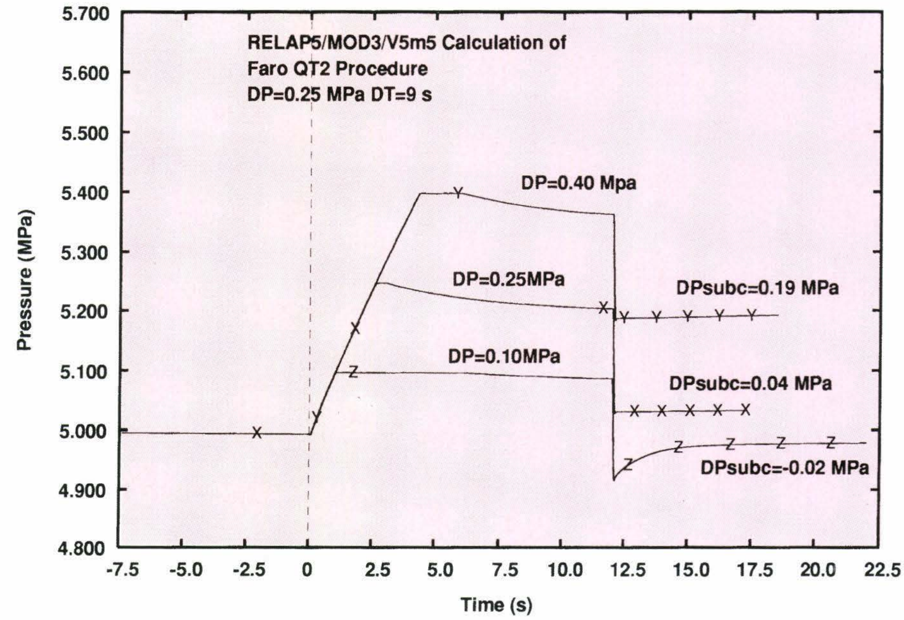

Fig. 1.63 Effect of initial TERMOS pressurization for the second quenching test of FARO facility

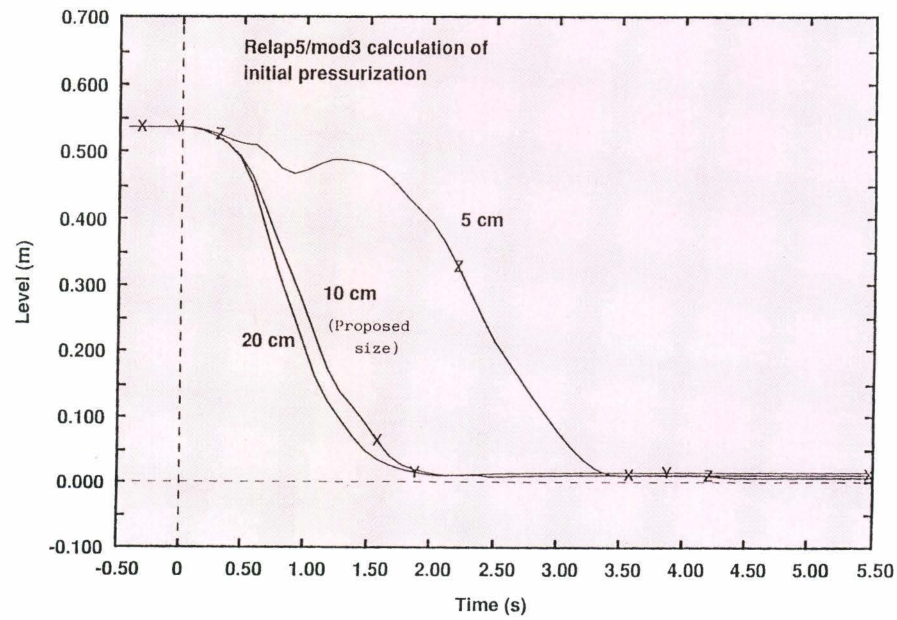

Fig. 1.64 Melt level during base case test simulation with 3 orifice diameters for the equalization valve

KROTOS Tests

In support to the large scale FARO tests the KROTOS facility was used during 1991/92 for FCI studies in the molten aluminum-oxide/water system. The objectives of these tests were to investigate in a 1D geometry the premixing of molten fuel jets in saturated and subcooled water, and to study the energetics of triggered or spontaneous steam explosions.

The experimental set-up and procedures were aleady described in the Annual Report 1991 (see also [12]), as well as the test results for the Al_2O_3/water experiments in nearly saturated water conditions (KROTOS 27 and 28). In both tests, melt penetration and mixing in the KROTOS test tube could be achieved. The externally triggered steam explosion in KROTOS 28 showed a high propagation velocity and super-critical pressures in the thermal detonation wave. In KROTOS 27, without external trigger, the melt was quenched and collected on the test tube bottom plate and only long-time boiling and melt-cooldown occurred. Due to the interesting results, the test series was continued studying especially the effect of water subcooling on the melt quenching behaviour.

The general arrangement of the KROTOS test section is again given in *Fig. 1.65* (here for KROTOS 26) which also shows the experimental instrumentation, mainly consisting of pressure transducers and thermocouples. In all the five tests (KROTOS 26 to 30) about 1.5 kg of Al_2O_3 melt (density 2.6 g/cm^3) at temperature 2300-2400°C was contained in a Mo crucible and dropped from the furnace onto a puncher where the 0.2 mm thick bottom membrane was ruptured (*Fig. 1.66*). The

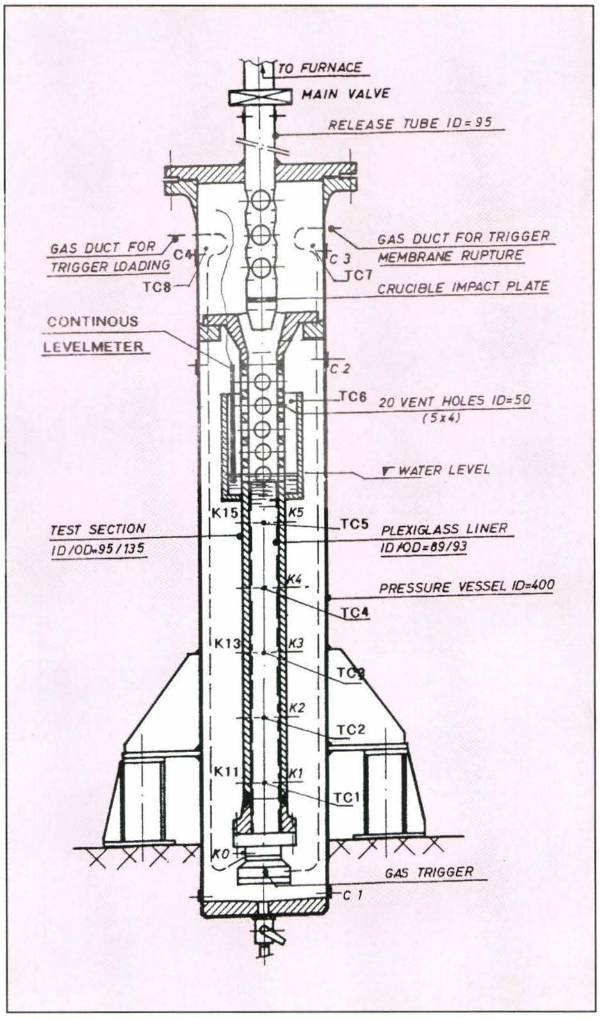

Fig. 1.65 KROTOS 26 experimental instrumentation

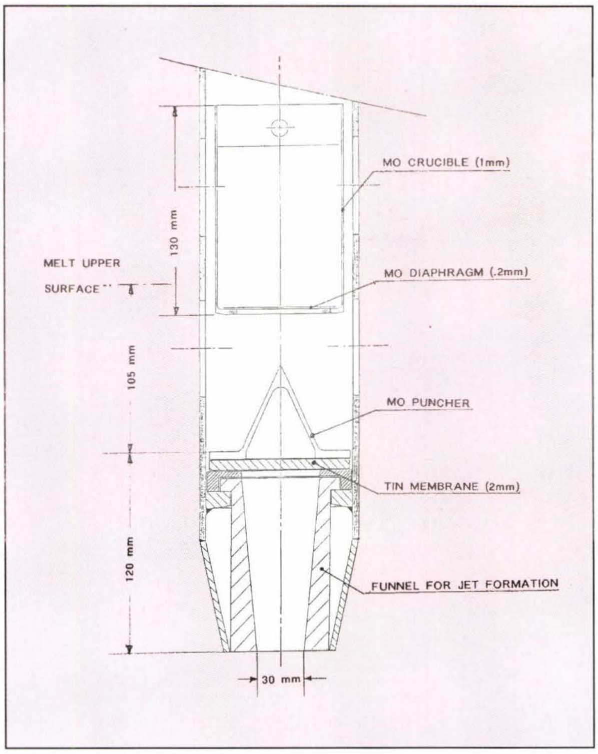

Fig. 1.66 Crucible with puncher for aluminuim oxide-water tests

release of the melt is delayed by some hundreds of ms by a tin disc placed below the puncher which has to be melted by the melt itself. In this way the melt is released by gravity through a nozzle which defines the 30 mm melt jet diameter. The nozzle exit was situated 0.46 m above the water free surface in the test tube. The initial water temperature was 90°C in KROTOS 27 and 28, 60°C in KROTOS 26, and 20°C in KROTOS 29 and 30, respectively. The tests were performed at an initial system pressure of 0.1 MPa. The height of the water column in the test tube varied between 1.08 and 1.12 m; the water column diameter was 89 and 95 mm, respectively.

In the following, main emphasis is placed on the tests performed under subcooled water conditions, i.e KROTOS 26, 29 and 30. Only in KROTOS 26 an external trigger system (gas trigger) was applied. The zero time signal for triggering and the data acquisition was generated by the falling crucible fracturing a copper wire in the release tube.

1. KROTOS 26 - 40 K Subcooled Water

The objective of this test was the triggering of a steam explosion at 0.1 MPa system pressure. From a mass of 1.4 kg Al_2O_3 at 2300°C, less than 1 kg was drained from the Mo-crucible. The trigger pulse was fired 2 s after 0-time but at that instant the melt front was at most above K4. Nevertheless, a steam explosion did occur.

A comparison of the transient pressure signals at locations K1 to K5 is shown in ***Fig. 1.67*** and may help in understanding KROTOS 26. At the locations K1 up to K3 the trigger pulse propagation is clearly seen, followed by a strong "recoil" pressure starting from the test tube upper region and propagating downwards, with a speed of about 1500 m/s, which corresponds to the speed of sound in water. At location K5 the trigger pulse was no longer visible and the K5 pressure rise to a level of about 25 MPa indicates that a steam explosion was initiated in this region.

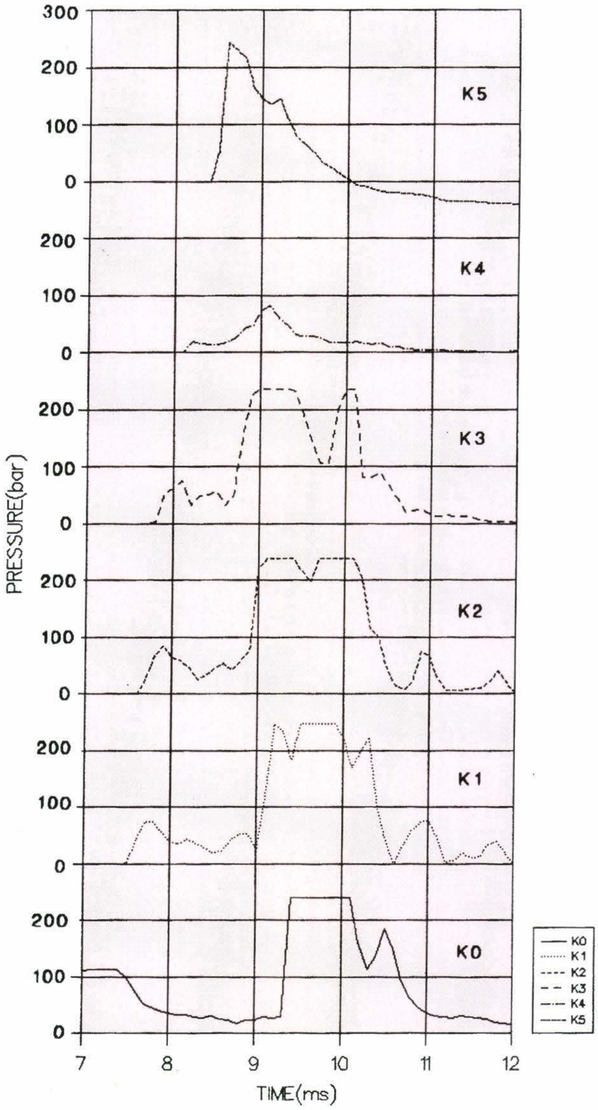

Fig. 1.67 KROTOS 26 - pressure signals K0 to K5

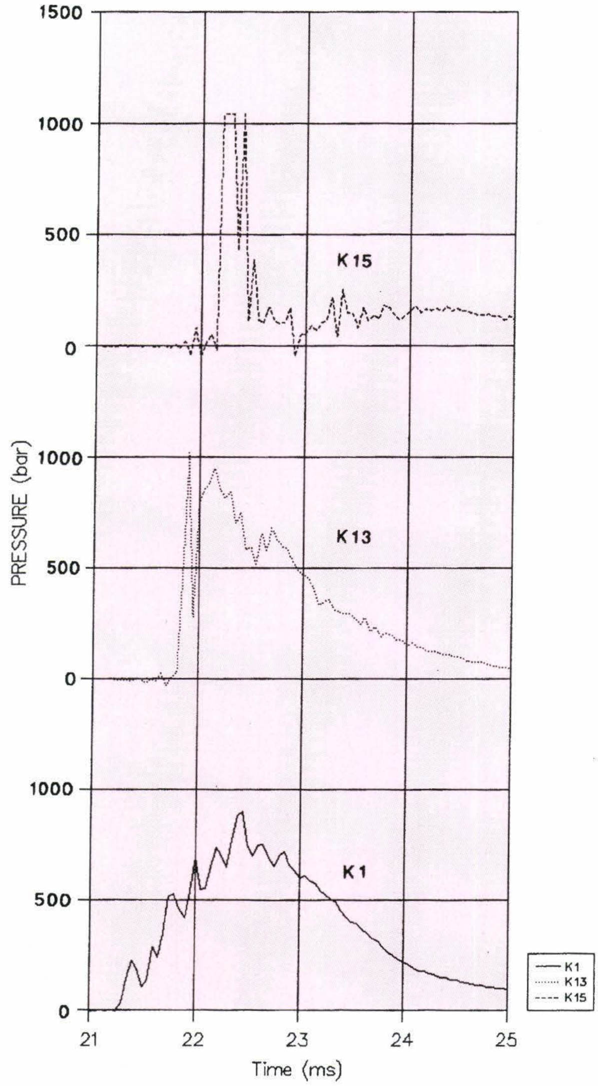

Fig. 1.68 KROTOS 29 - pressure signals K1, K13 and K15

2. KROTOS 29 - 80 K Subcooled Water

The objective of test KROTOS 29 was to investigate the quenching behaviour of an Al_2O_3 melt in highly subcooled water at 0.1 MPa system pressure. Triggering off a steam explosion was not intended, therefore the trigger system was not included. 1.5 kg molten Al_2O_3 at 2300°C was released into the test tube filled with water at 20°C. The total melt mass entered the water in approximately 2 s.

The melt leading edge had penetrated down to level K1 (15 cm above bottom plate), when a self-triggered steam explosion occurred at 2.95 s after breaking the copper wire. The melt-water mixing region was similar to KROTOS 28 and explosion pressures up to 100 MPa were measured (*Fig.1.68*).

In the pressure vessel an additional pressurization of about 0.2 MPa with respect to the initial pressure was recorded.

3. KROTOS 30 - 80 K Subcooled Water

This test was a repetition of KROTOS 29. However, an attempt was made to avoid self-triggering of a steam explosion introducing some modifications in the test arrangement, e.g. the tin membrane below the crucible had been removed, in order to inhibit the generation of an explosion due to a possible tin/water interaction ahead of the Al_2O_3 melt jet. 1.5 kg molten Al_2O_3 at 2300°C was released into the test tube filled with water at 20°C. The melt left the nozzle exit (no tin disc) with a much higher velocity than in tests KROTOS 26 to 29. For the KROTOS 30 test arrangement, an initial melt release velocity of the order of 25 m/s was deduced from simulation experiments. In KROTOS 30 all the melt entered the water. At 1.3 s, after breaking the copper wire, when the melt was still above level K3 a very strong self-triggered explosion occurred. The transient pressure signals showed that the explosion propagated downwards from the top of the test tube and led to exceptionally high pressures in the lower test tube region where only water was present (see KROTOS 26). Pressures of 90 MPa were recorded at level K5 and more than 100 MPa at the lower levels. In the cover gas of the pressure vessel an additional pressure of 0.55 MPa with respect to the initial pressure was measured 30 ms after the explosion.

The four pressure transducers mounted in the lower part (pure water region) of the test tube were blown out from the wall and damaged by striking the outer pressure vessel, causing internal indentations (*Figs. 1.69* and *1.70*). Due to the very strong water hammer the twelve bottom plate bolts (M 16) were plastically elongated up to 6 mm and the test tube diameter was distended by about 1%. It was estimated that the impact pressure must have been at least 150 MPa. The value of the impulse which depends on the duration of the pulse could not be calculated because the pressure gauges were destroyed early. As in KROTOS 26, 28 and 29 the upper water container and level-meters were destroyed. Post-test analysis showed an extremely fine fragmentation of the melt, never found in our previous experiments.

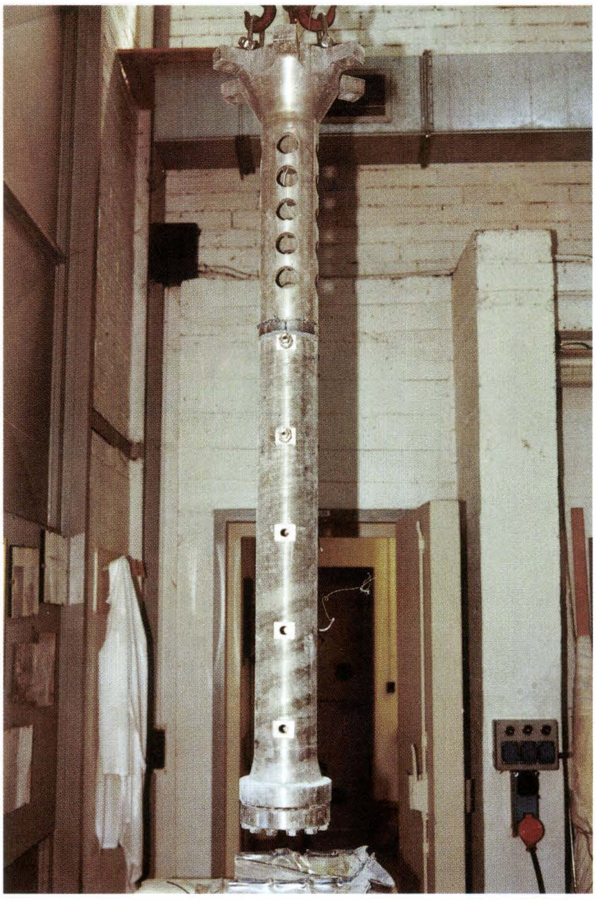

Fig. 1.69 KROTOS 30: view of test tube

Table 1.6 summarises the experimental results of the five Al_2O_3-water tests performed. More details can be found in [13]. The generation of stable coarse mixture was only possible under nearly saturated conditions (KROTOS 27,28), and in order to produce a steam explosion an external trigger was needed, as verified in test KROTOS 28. No noticeable difference in the melt penetration was found in neither nearly saturated nor subcooled water. Comparing the tests KROTOS 28 (10 K subcooled) and 29 (80 K subcooled) the melt leading edge velocities were similar (KROTOS 28: 0.6 m/s, KROTOS 29: 0.5 m/s).

On the basis of the present KROTOS test series and of our past work in the system Sn/H_2O, it is reasonable to think that transition to unstable film boiling, when the coarse mixture contacted the structures, acted as a trigger in KROTOS 29. For KROTOS 30 it is thought that the enhanced fuel breakup, induced by the higher velocity melt release into the water, facilitated the spontaneous triggering of the explosion.

In the tests KROTOS 26, 28 to 30 supercritical steam explosions were encountered, detonation wave propagation velocities of 650-1000 m/s were measured in the melt/water mixture. In KROTOS 26 and KROTOS 30 the explosion occurred in the upper part of the test tube. Consequently, the water in the lower region remained single-phase and the base plate was subjected to a strong water hammer.

Fig. 1.71 shows the particle size distributions for the tests KROTOS 27 to 30. It is clear that KROTOS 27, in which no steam explosion happened, gives a small percentage (10%) of particles smaller than 6 mm. KROTOS 28 and 29 show nearly the same particle size distribution with a large percentage of small particles (about 60% smaller than 250 μm). The debris in KROTOS 30 was extremely fine (85% below 250 μm).

During the experiments there was a pressurization of the pressure vessel. After an initial pressure peak caused by the rapid steaming which immediately followed the explosions, the pressure remains relatively constant for some time. In KROTOS 30, the additional pressure in the pressure vessel was 0.55 MPa, the largest one found in all tests. An estimate of the explosion work done by assuming an isentropic compression of the cover gas gives 1.3,0.8 and 1.25%, respectively for KROTOS 28, 29 and 30. This is only a lower bound estimate, and does not include mechanical work plastically deforming the various components in the pressure vessel. These conversion ratios are related to the interacting fuel mass which was fragmented in particle sizes less than 250 μm.

The Al_2O_3 water mixing test series gave interesting results, especially in view of the high pressures found in the explosions. The results can be used to verify FCI fragmentation models in thermodynamic codes, not only for the coarse mixing phase, but also for the thermal detonation phase. Further KROTOS tests are envisaged using prototypic reactor melt materials, that is, mixtures of 80 w% UO_2-20 w% ZrO_2, with melt masses of up to 5 kg.

Fig. 1.70 KROTOS 30: view of test tube bottom plate

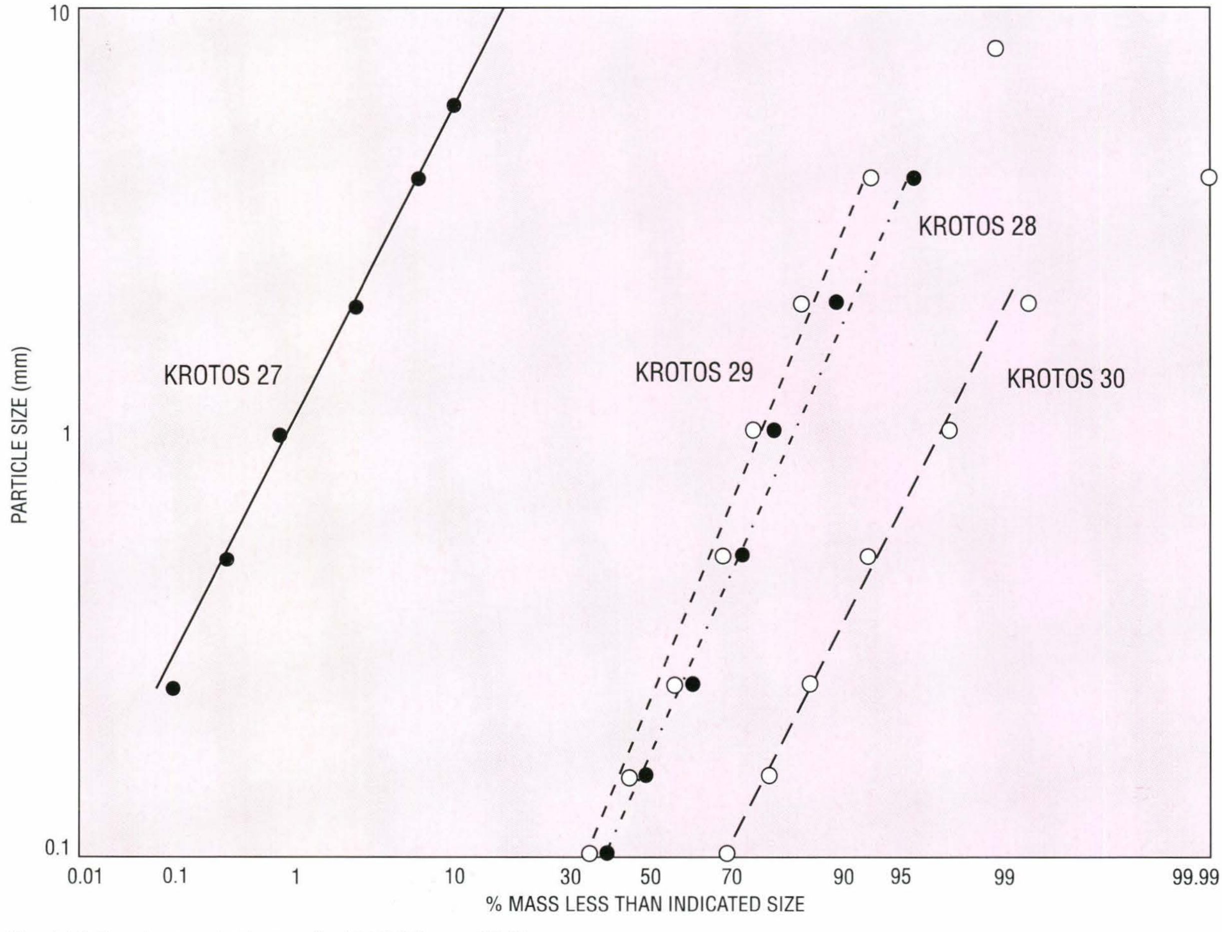

Fig. 1.71 Particle size distributions for KROTOS tests 27-30

Table 1.6 Summary of KROTOS experimental results

KROTOS	27	28	26	29	30
Tsub, K	10	10	40	80	80
Water temperature, °C	90	87	60	20	18
Tin-disc, mm	2	2	7	2	2
Mixing region	full tube	K1/K2	top K4/K5	K1/K2	top K3/K4
Leading edge velocity, m/s	0.4	0.6	n.a.*	0.5	0.8
Penetration depth, cm	full tube	90	35	90	40
External trigger	no	yes	yes	no	no
Loading pressure, MPa	–	8.5	11.2	–	–
Trigger time, ms	–	2528	2000	–	–
Spontaneous trigger	–	–	–	yes	yes
Trigger time, ms	–	–	–	2950	1305
Propagation velocity, m/s	–	650	n.a.*	1000	1000
Explosion pressure, MPa	–	>50	>25	>100	>100
Additional chamber pressure, MPa	small	0.25?	0.05	0.18	0.55
Total debris, g	1100	1215	950	1350	1400
Debris, smaller than 100 μm, g	–	480	180	560	1000

* n.a. - not available

Analysis of KROTOS experiments with TEXAS-III

A new version of the computer code TEXAS, named version III, to deal with vapour explosions, has been developed through a collaboration contract with the University of Wisconsin. The conceptual picture of the model adopted to describe the vapour explosion in TEXAS-III, is represented by vapour film collapse due to Reyleigh-Taylor instabilities generated by a "trigger" pressure pulse and subsequent penetration of coolant microjets into the molten fuel. The rapid vapourization of these microjets within the fuel drops, produces local pressure growth and fuel fragmentation thus increasing the fuel surface area of heat exchange and hence vaporizing more coolant with subsequent additional pressure increase. This mechanism enhances the process of film collapse in the surrounding fuel drops, leading to a spatial explosive-like propagation phenomenon.

The code TEXAS-III has been used to analyse the test KROTOS 21 in which 7.5 kg of tin at 1370 K have been released into water at 361 K. The water and the free board volume have been nodalized into 12 and 8 Eulerian cells, respectively, and the melt has been partitioned into 40 Lagrangian parcels entering the water at a uniform rate. At 0.8 s after the melt release, the tin has almost reached the bottom of the KROTOS vessel and the predicted vapour void fraction is relatively large (15%-25%) because the subcooling of the water is low (12K).

The trigger is simulated by the expansion of a high pressure/small volume of steam (12 MPa, 15 cm^3) about 1 s after the penetration of the melt into the water. *Fig. 1.72* shows the pressure time histories in the Eulerian cells corresponding to the pressure transducers K0, K1, ..., K5. The peak pressures are well predicted but the propagation velocity is a little slower. The pressure rise in the gauges K0 to K2 from 8 s to 12 s have not been well predicted probably because the trigger has been modelled by

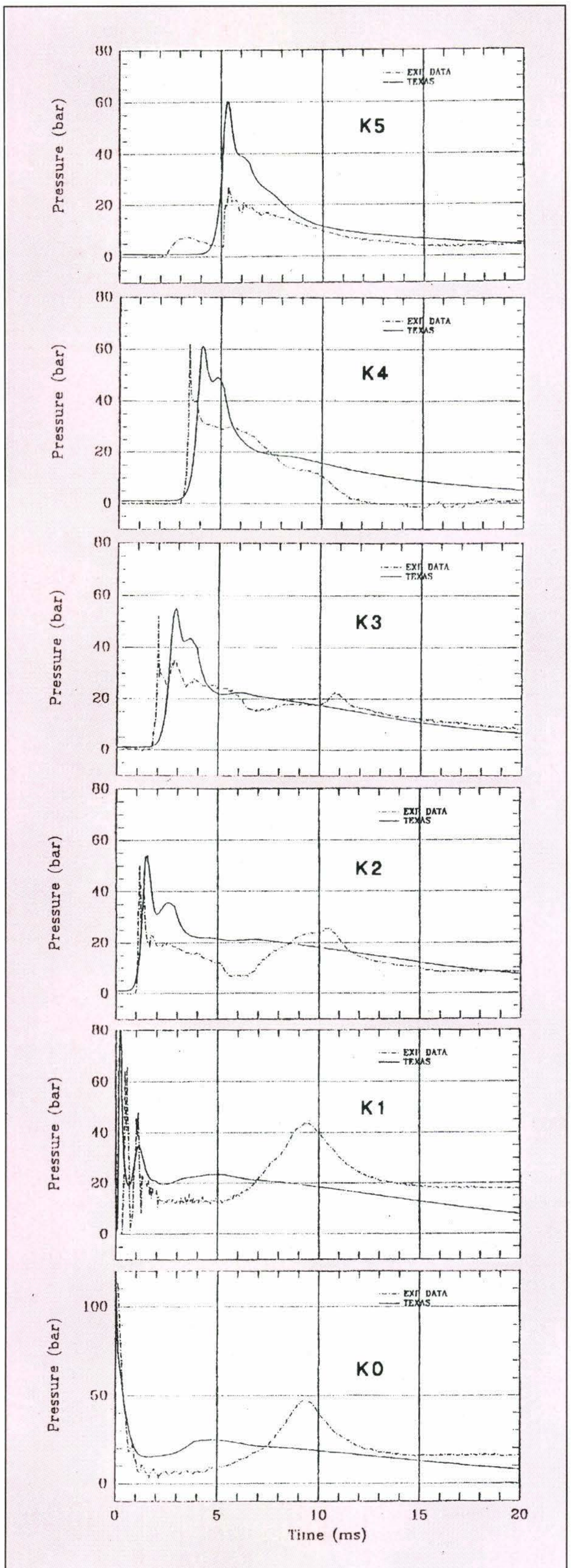

Fig. 1.72 KROTOS 21 - pressure signals K0-K5

steam and not as a non-condensable gas, like in the real test. Further improvements of the vapour explosion model in TEXAS-III are foreseen, including the effect of non-condensable gases.

KROTOS Analysis using a Thermodynamic Model

An assessment of the explosion pressure and propagation velocity without a spatially detailed knowledge of fuel-coolant mixing and energy transfer rate can be obtained by applying the steady-state shock adiabatic model which was originally proposed by Board and Hall. This model was developed by applying the classical theory of detonation in chemically reactive flows to the one dimensional case of an explosion front propagating through a coarsely mixed region of fuel and coolant.

With reference to a planar explosion front which is steadily progressing through uniformly mixed materials initially at rest, and leaving behind an equilibrium mixture of "exploded" materials, by applying the equations of conservation of mass, momentum and energy, the Hugoniot relationship can be derived which defines a unique relationship between the possible values of pressure and specific volume, p_2 and v_2:

$$\frac{1}{2}(p_1+p_2)(v_1-v_2)=I_2-I_1 \qquad (1)$$

where I is the specific internal energy before (1) and after (2) the shock. The velocities u_1 and u_2 in the moving frame of reference of the shock front are given by:

$$u_1=v_1\sqrt{\frac{p_2-p_1}{v_1-v_2}} \qquad (2)$$

$$u_2=v_2\sqrt{\frac{p_2-p_1}{v_1-v_2}} \qquad (3)$$

u_1 corresponds to the shock propagation velocity. The velocity of the material behind the shock front in the laboratory frame is u_1-u_2. Therefore, the medium behind the shock front moves in the direction of the front. Accordingly, the region behind the front must be followed by a region of expansion, which may be thought of as providing the driving power for the explosion. The leading edge of this expansion region is a rarefaction wave which travels at the local mixture sound speed, c_2; for this to be stationary in the frame of shock front u_2 must be equal c_2 (Chapman-Jouguet condition). This condition defines a unique equilibrium state on the detonation branch of the Hugoniot curve (Eq. 1).

Such a model can be used to estimate the mixing conditions prior to an explosion based on the measured explosion pressures and propagation velocities. *Figs. 1.73* and *1.74* have been derived from a calculation using equations (1)-(3) for a tin-water system corresponding to the initial temperatures for KROTOS-21. Note that the two independent variables are the initial mixture void fraction and the mass ratio of coolant to fuel. A specific pair of these independent variables provide a unique prediction of the C-J propagation pressure and shock velocity for the experiment.

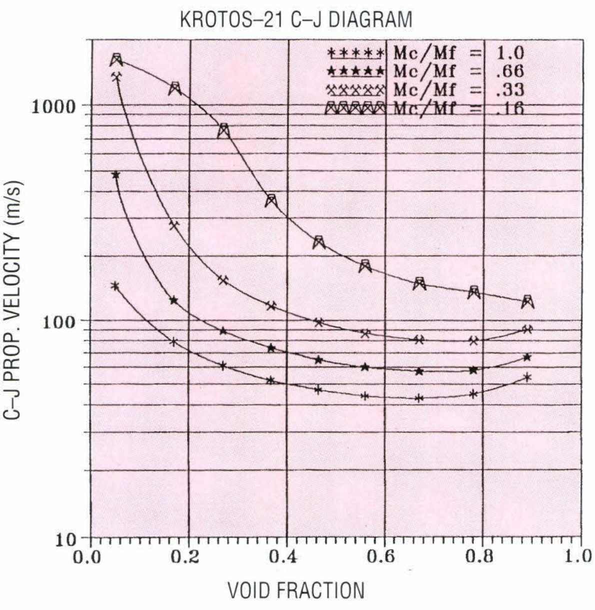

Fig. 1.73 KROTOS 21: C-J prop. velocity diagram

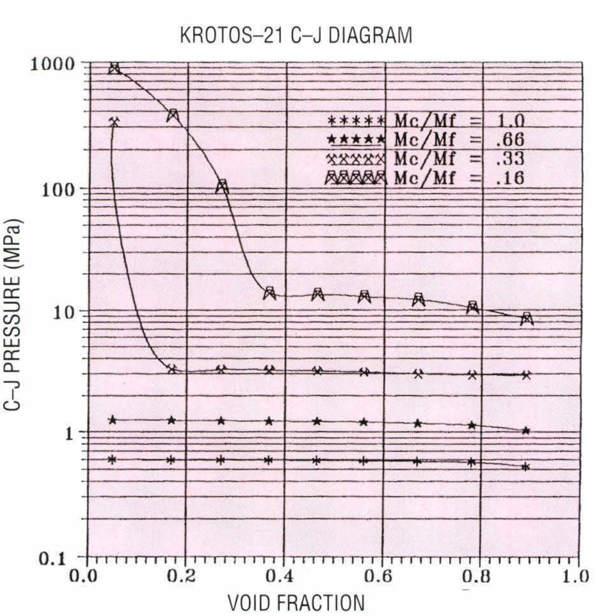

Fig. 1.74 KROTOS 21: C-J pressure diagram

Based on the data for KROTOS-21, the C-J pressure was about 3.5-4.0 MPa with an average propagation velocity of 200-225 m/s. Using these results, the average mixture conditions are 0.27-0.55-0.18 for the fuel, coolant liquid and vapour volume fraction respectively. This provides the experimentalist with a way to estimate the likely conditions within the mixture prior to the explosion. These "explosion propagation" curves can also be used to examine the effect of the initial conditions on the explosion.

References

[1] MAGALLON D. et al. - Scoping Test Data Report - JRC-Ispra, Technical Note No I.92.135, December 1992

[2] WIDER H. et al - Quick Look Report on the Scoping Test - JRC-Ispra, Technical Note No I.92.139, December 1992

[3] MAGALLON D., HOHMANN H. - High Pressure Corium Melt Quenching Tests in FARO - CSNI Specialist Meeting on Fuel Coolant Interaction, Santa Barbara (CA), 5-8 Jan.1993

[4] WANG S.K., BLOMQUIST C.A., SPENCER B.W., MAC UMBER L.M., SCHNEIDER J.P. - Experimenal Study of the Fragmentaion and Quench Behaviour of Corium Melts in Water - Nat. Heat Transfer Conf., ANS, 1989

[5] MAGALLON D., ZEYEN R., HOHMANN H. - 100 kg-scale Molten UO_2 Out-of-Pile Interactions with LMFBR Structures: Plate Erosion and Fuel Freezing in Channels - Int. Conf. on Fast Reactor Core and Fuel Structural Behaviour held in Inverness on 4-6 June 1990, BNES

[6] CHU C.C., CORRADINI M.L., MURPHY J., TANG J. - A Code Manual for TEXAS-II - Nuclear Engineering and Engineering Physics Dept. Univ. of Wisconsin - Madison WI 53706, May 1991

[7] YOUNG M.F., GIGUERE P.T. - The IFCI (Integrated Fuel-Coolant Interaction) Code: Models, Correlations and Quality Assurance - Sandia Innovative Technology Applications Division, Sandia National Laboratories, Albuquerque, NM 87185, 1990

[8] DOSANJH S.S., ed. - MELPROG-PWR:MOD1: A Two-Dimensional, Mechanistic Code for Analysis of Reactor Core Melt Progression and Vessel Attack Under Severe Accident Conditions - SAND88-1824, NUREG/CR-5193, Albuquerque, N.M., 1989

[9] CARLSON K.E., et al - RELAP5/MOD3 Code Manual - Draft Volumes 1,2,3 and 4, NUREG/CR-5273, EGG-2596, June 1990

[10] ANNUNZIATO A. - Pre-Test Analysis of FARO Quenching Test 2 - Presentation at JRC Ispra, internal communication, 23.04.1992

[11] ANNUNZIATO A. - Analysis of the FARO facility inital pressurization behaviour during the 150 kg test - Technical Note, 09.12.1992

[12] HOHMANN H. et al. - KROTOS 26 to KROTOS 30: Experimental Data Collection - JRC-Ispra, Technical Note No. I.92.115, November 1992

[13] HOHMANN H. et al. - Experiments in the Aluminumoxide/Water System - CSNI Specialist Meeting on Fuel Coolant Interaction, Santa Barbara (CA), 5-8 Jan. 1993

1.1.3 THERMALHYDRAULICS

The activities in the field of reactor thermal-hydraulic safety research and development have been concentrated on the analysis and documentation of the LOBI test data and on the assessment and improvement of the LWR safety code CATHARE.

LOBI

A comprehensive experimental data base, pertinent to a wide range of postulated accident conditions in Pressurized Water Reactors (PWRs) has been established in the framework of the LOBI (LWR Off-normal Behaviour Investigation) experimental programme carried-out from December 1979 to June 1991. In its final form, the LOBI Data Base comprises the test data of 70 experiments performed in the LOBI test facility, a 1:700 scale model of a 4-loop 1300 MWe PWR.

As for the programmatic objectives of the LOBI research programme, the acquired data base provides a valuable set of reference information for the evaluation of PWRs loss-of-coolant accidents (LOCAs) and other risk dominant sequences such as anticipated as well as abnormal transients. It is of specific relevance for the:

- identification and/or verification of basic thermal-hydraulic phenomenologies governing the evolution of PWR accidents with emphasis on the performance of the engineered safety systems and the effectiveness of accident management strategies,
- development and/or validation of analytical models and the assessment and/or improvement of the predictive capabilities of system codes used in water cooled reactors safety analysis.

The LOBI test data are an integral part of the reactor safety research and code assessment programmes of several EC member countries. LOBI data are systematically used being included in the assessment matrices of the French code CATHARE and of the German code ATHLET. Various research organisations are also using the data for the assessment of other system codes such as RELAP.

Through a special arrangement with the Organisation for Economic Cooperation and Development (OECD), some LOBI tests have been included in the international code assessment programme of the OECD Committee on the Safety of Nuclear Installations (CSNI) and are thus available also outside the EC context.

Some research organisations from Central- and East-European countries have requested the access to and the use of the LOBI Data Base. In view of the emerging cooperation in science and technology with non-EC member countries, the data are being released following the establishment of collaborative agreements in the frame of Article 10 of the EURATOM Treaty.

During the reporting period a considerable effort has been devoted to data analysis and documentation to provide prospective users with the required reference information for their interpretation as well as to data base maintenance and users service. The data reports and the analysis reports of the following tests (*Table 1.7*) have been finalized and issued.

The assessment of large system codes used in the safety analysis of water cooled reactors is generally

Table 1.7 LOBI data and analysis reports (1992)

Data Reports	Analysis Reports
BL-34: Small 6% CL Break LOCA at Low Power [1]	***BL-16:*** Small 0.4% Cold Leg Break LOCA with Asymmetric Cooldown of the Secondary System [5]
BL-44: Small 6% Cold Leg Break LOCA at High Power [2]	***BT-17:*** Loss of Feed Water (LOFW) with Secondary System Feed and Bleed [6]
BT-17: Loss of Feed Water (LOFW) with Secondary System Feed and Bleed [3]	***BL-44:*** Small 6% Cold Leg Break LOCA at High Power [7]
BL-40: Steam Generator Tube Rupture (SGTR) in a 1-Loop PWR [4]	

based on test data from scaled integral system and/or separate effect test facilities. Relevant test data from the reference reactors would be certainly desirable. Clearly, the acquisition of experimental data in a full size plant and especially under accident conditions is prohibitive for obvious economic and practical considerations.

The typicallity of experimental data acquired in test facilities is, however, often questioned due to inherent scaling distortions or simulation compromises. Controversy thus arises when the predictive capabilities of a safety code are extrapolated from a scaled installation to the full-size plant. To lend credence to the adequacy of system codes in predicting a realistic system response, the assessment process is diversified to a wide range of test data from different scaled test facilities.

Within this context, of particular significance are the results of Test BL-34 which represent the small break LOCA test performed in the framework of the counter part test (CPT) programme agreed upon with CEA (France), JAERI (Japan) and SIET (Italy). Similar tests have been performed in the BETHSY, LSTF and SPES test facilities characterised by different scaling ratios and/or design concepts. A comparative analysis of the test results [8] indicates the occurrence of similar major phenomenologies in the four test facilities, *Figs. 1.75* and *1.76*. The relevant fluid distribution within the LOBI installation during the early phase of the transient is illustrated in *Fig. 1.77*.

Accident management strategies alternative to conventional emergency operating procedures are being proposed to enhance the safety margins of reactors of current design or are being integrated in the conceptual development of advanced reactor configurations. The effectiveness of secondary system feed and bleed in the LOBI and in the SPES test facilities has been assessed through a comparative analysis of the results from similar tests [9, 10].

In line with the tradition of an effective approach to international cooperation in the field of water cooled reactor safety research and upon the specific request from several EC research organisations, the LOBI Data Users Club has been established and convened. The terms of reference of such a club is to provide the framework for a continuing collaboration

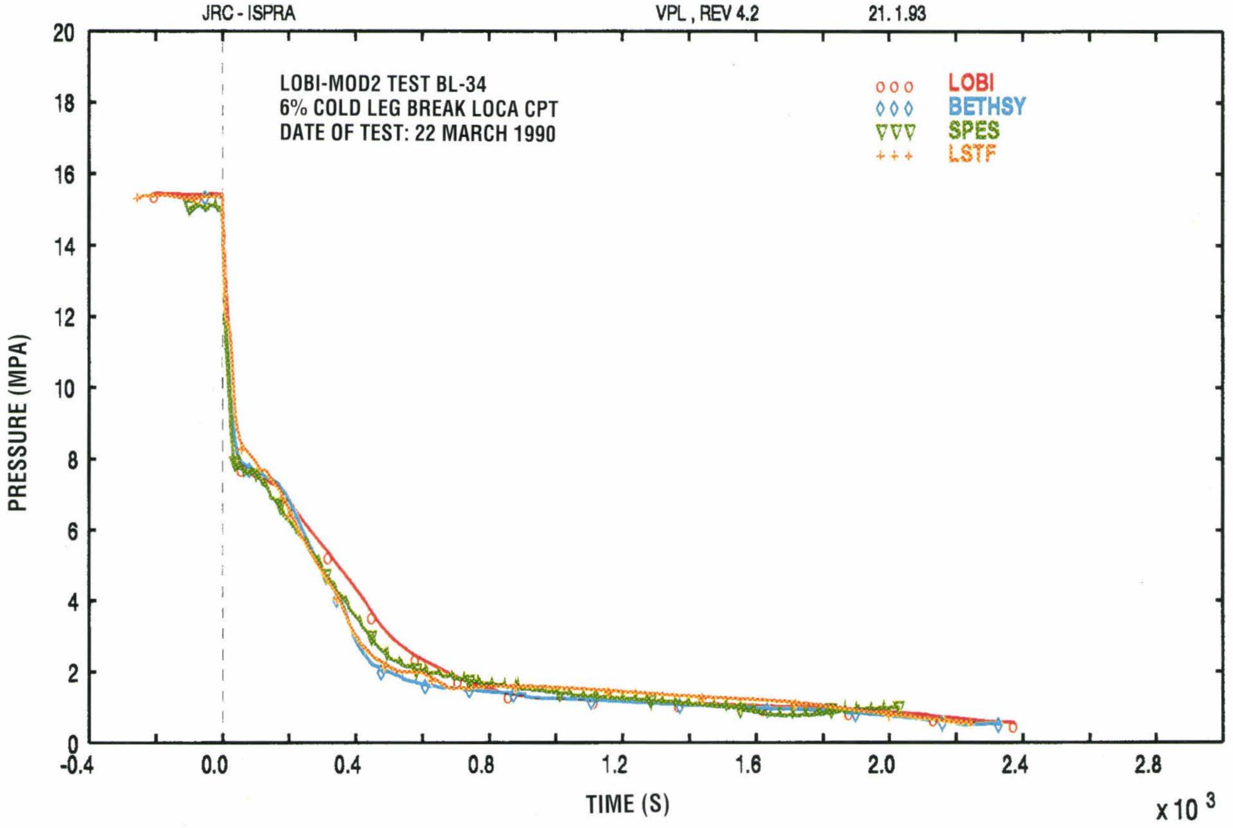

Fig. 1.75 Primary system pressure response in the LOBI, SPES, LSTF and BETHSY counter part test

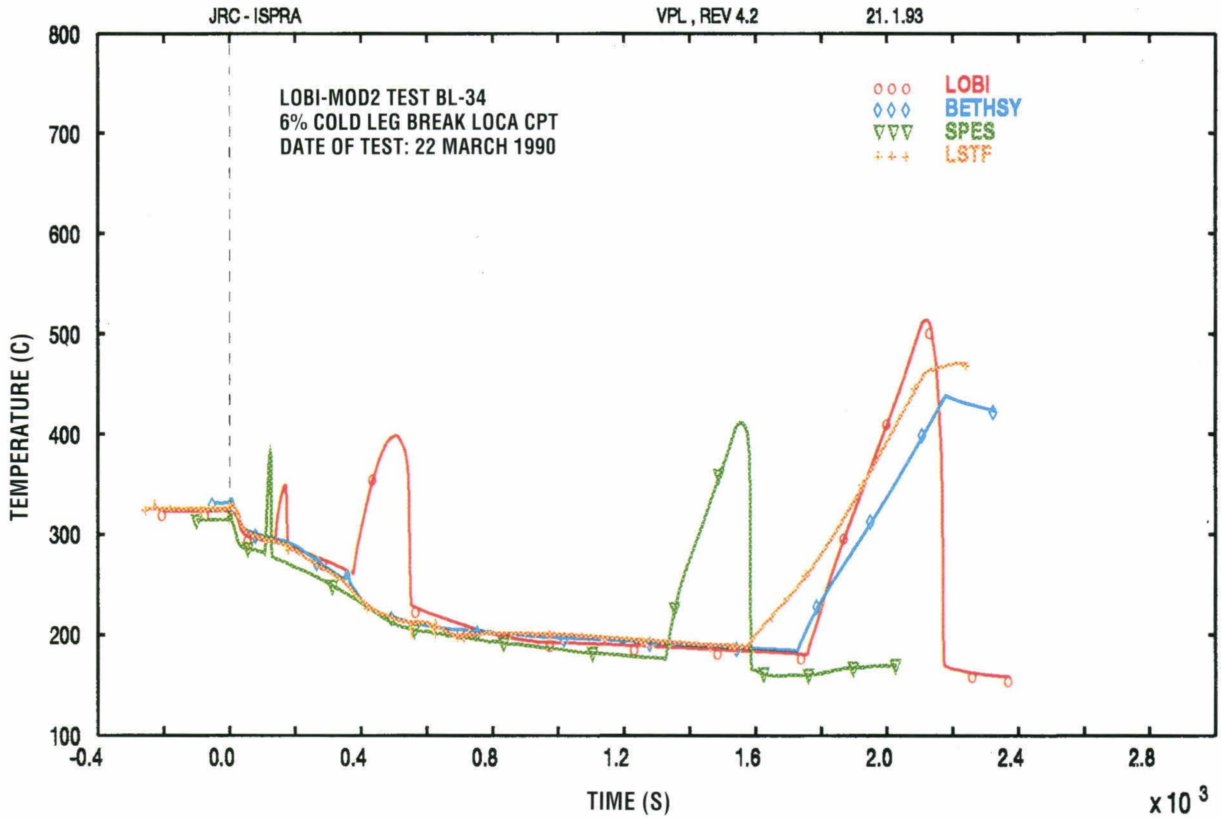

Fig. 1.76 Core temperature response in the LOBI, SPES, LSTF and BETHSY counter part test

among the experts from institutional and/or industrial organizations of EC member countries (*Table 1.8*) which have previously contributed to test definition and specification and are now actively involved in post-test analysis of the results.

A LOBI Seminar has been organised to provide an open forum for the presentation of major achievements and significant results of the research activities. The Seminar which was held from March 31st to April 2nd, 1992, in Arona (I) convened about 60 experts from industrial and institutional organisations of EC member countries, EFTA as well as Central and East-European countries and international organisations such as the IAEA. The Seminar offered an opportunity to scientists actively involved in the LOBI test definition and specification as well as in the pre- and post-test analysis of the results to summarize their related research activities [11 to 15]. It also offered an opportunity for an international exchange of technical and scientific information on current requirements in water reactor safety research.

Table 1.8 LOBI data users club

Country	Institutions
Germany	GRS, Siemens-KWU, Battelle
France	CEA, FRAMATOME
U.K.	Nuclear Electric, AEA Technology, Rolls Royce
Belgium	TRACTEBEL
Italy	ENEA, Pisa University
Spain	CIEMAT, UPC, Union FENOSA
Hungary	KFKI Atomic Energy Research Institute
Poland	Institute of Atomic Energy
Russia	Science and Engineering Safety Centre Electrogorsk Research and Engineering Centre

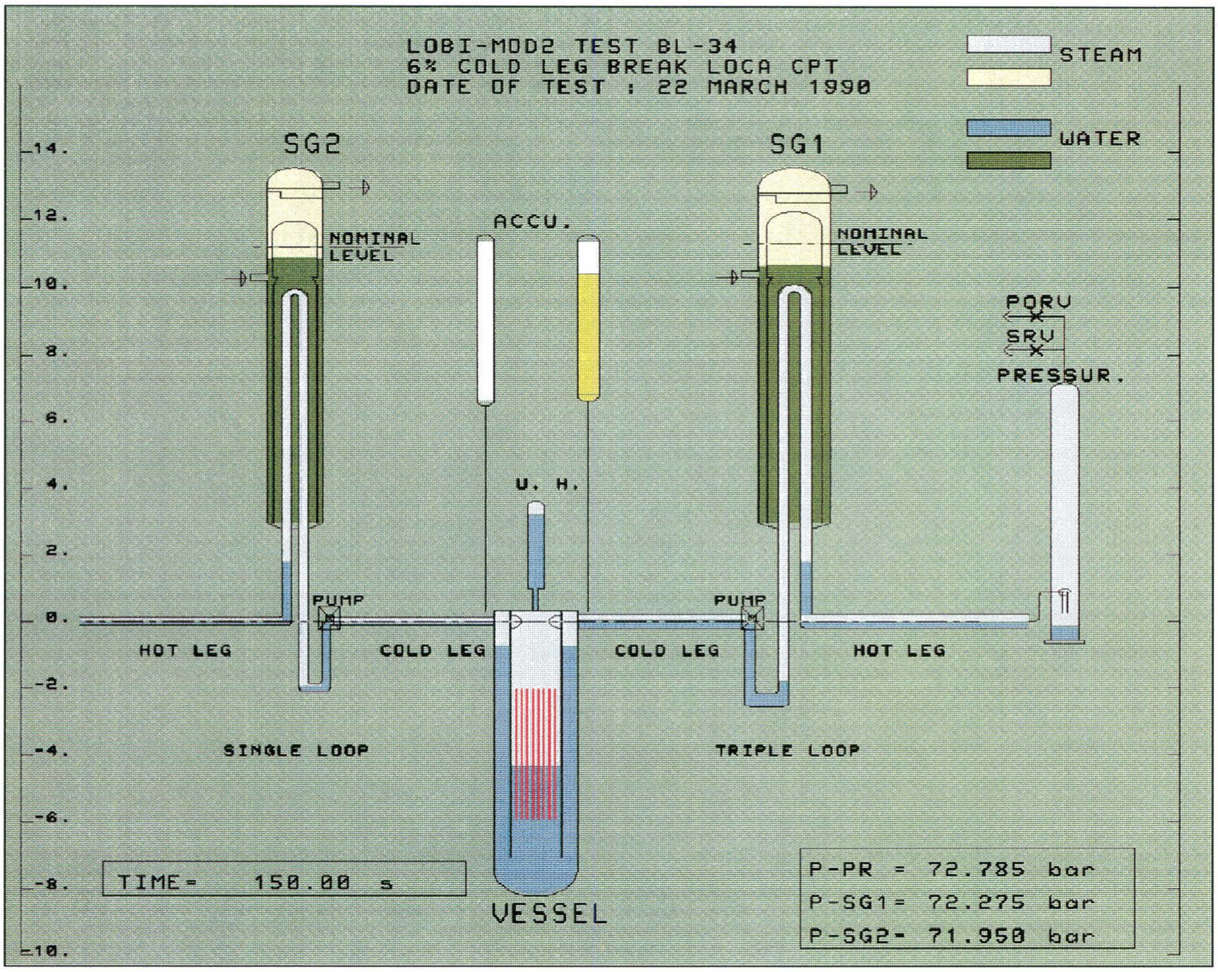

Fig. 1.77 Fluid distribution in the LOBI counter part test at 150s after start of LOCA

Assessment and Improvement of LWR Safety Code CATHARE2

During the period 1990-92, the JRC-Ispra has collaborated with the Centre d'Etudes Nucléaires de Grenoble (CENG) on the independent assessment of the LWR safety code CATHARE2 in the framework of a 3-years bilateral agreement between the JRC and the CEA. This activity terminated at the end of 1992 with the completion of the final set of calculations performed with the JRC Amdahl.

The contract between JRC and CENG, foresaw the use of experimental data from the LOBI test facility for evaluation of the code against various prototypical reactor accident simulations. Calculations defined for code comparison included five LOCA experiments from the LOBI test facility - two 1% cold leg break LOCAs (Tests A2-81, BL-12), a 6% low-power LOCA experiment (Test BL-34) and two steam-line break secondary-side LOCAs (Tests BT-01, BT-12). These have now been fully documented [16-21] and a final report [22] on the JRC code assessment activities has been completed. Results of these calculations, reported at the regular CATHARE Users Club meetings [23-25] held twice yearly in Grenoble, and various specialist conferences [26-28], indicate some modelling problems, the rectification of which would lead to much better agreement with measured data for both primary and secondary side behaviour. Difficulties encountered in installing CATHARE2 to run under UNIX environment have also been reported [29,30].

The CATHARE2 code, developed entirely by CENG, is widely used throughout Europe as a best-estimate LWR safety code for thermal-hydraulics analysis. It is currently the subject of an intensive programme of assessment by various institutes world-wide, as indicated by the large attendance at the 4th CATHARE-BETHSY International Seminar [26]. The latest version of the code (Version 1.2e, Rev. 5), was released in early 1991 for general use among the utilities for safety studies and for code qualification purposes. It demonstrates significant improvements over earlier versions of the code, first released in September 1979 as CATHARE1.

CATHARE2 is a 6-equation (2 mass, 2 momentum, 2 energy) non-equilibrium inhomogeneous two-fluid (steam and water plus non-condensable gases) systems code for studying transient thermal hydraulic phenomena in integral systems such as PWR primary and secondary cooling circuits. The code is structured around the principles of object-oriented programming, using modules (objects) to build up a complex network of interacting components (i.e. pumps, pipes, vessels, T-junctions, annuli, etc.). It comprises 1 or 2 mass-balance equations for the transport of non-condensable gases (H2, N2), boron and/or fission products. A fully-implicit finite-difference scheme is used based on a donor-cell staggered-mesh approach, resulting in a strongly non-linear system of equations which is solved by a Newton-Raphson iterative procedure.

Code implementation

Version 1.2 was installed on the JRC Amdahl computer under MVS in April 1990, version 1.2e was installed in November, and further updates completed in December 1990. Modifications were made at the same time to various post-processing routines to interface CATHARE2 with existing and new post-processing software packages.
Version 1.2e/1 of CATHARE2 was installed under UNIX on three different workstations available at the JRC during 1991 - a DEC 5000/200, an IBM Risc 6000-530H and an HP 750. As explained in [29,30], the installation was straight forward on the first two workstations, but on the HP workstation the code had to be compiled with a special data alignment option which drastically degraded the computer's performance. This was investigated by the CATHARE maintenance team at Grenoble and the problem was eventually resolved to allow partial use of the code on the HP machine.

Post-test Calculation Results

A standardised code input dataset has been set up to model the LOBI test facility [16], and a number of LOBI experiments were selected in collaboration with CENG to help in assessing different aspects of the code.

LOBI Test A2-81

This was a 1% cold leg break LOCA from nominal conditions with scram, normal pump coast down, delayed HPIS injection to the intact loop cold leg and an imposed secondary-side cooldown rate of 100 K/hour. Calculations with CATHARE2 Ver.1.2e/2 Rev. 5 and Rev. 4, using the standard LOBI nodalisation, were made together with various sensitivity studies to check different aspects of the code and nodalisation arising from the different sets of physical closure laws in the two code versions. A full description of the test and associated calculations are given in [17].

Problems common to both calculations related to the unphysical prediction of water hold-up in the upper plenum which seriously affected the total mass inventory and caused unobserved loop seal clearing. Incorrect void distribution in the pressure vessel was traced to excessive interfacial friction between the two phases in the core region, coupled with high condensation rates near the ECCS injection point. Sensitivity studies localised the problem to inappropriate modelling of the vessel upper power connecting plate. A further study was made to look at voiding of the steam generator U-tubes. Code results indicated that if any liquid at all was present in the U-tubes, there would be no filmwise condensation which drastically reduced the condensate production and run-back of liquid into the loop seals. Sensitivity calculations exposed a heat transfer problem in the rising leg of the U-tubes during high pressure condensation.

LOBI Test BL-12

This test was a beyond-DBA 1% cold leg break LOCA scenario defined by the CEA as part of the LOBI B-test programme. It is one of a series of small break LOCA tests, along with A2-81, having complementary boundary conditions. This test had no HPIS injection and no secondary-side cooldown (only the natural cooldown rate), but it did have accumulator injection to the intact loop cold leg. A strong dryout at high pressure, with core temperatures rising to above 650°C, allows the relevant heat transfer models to be assessed.

The test has been calculated with CATHARE2 Ver.1.2e/2 Rev. 5 and Rev. 4, using the standard LOBI nodalisation. A full description of the test and associated calculations can be found in [18].

Specific modelling problems related to an incorrectly predicted dryout (caused by bulk boiling in the core instead of gradual exposure of the rods through liquid boiloff) and to incorrect modelling of condensation during accumulator injection. The former problem is due to a geometry-related influence at the core outlet which artificially increases the interfacial drag between the two phases, and to a 'hot wall' effect in the downcomer which feeds the core with near-saturated hot water. The increased interfacial shear prevents liquid from draining into the heated part of the core, so the pressure in the core and hot legs increases causing the loop seal to blow erroneously. The second problem was identified as being due to a missing hydrostatic gravity-head term in the accumulator model, compounded by practical difficulties in modelling the real dynamic behaviour of the accumulator feed line and isolation valve.

LOBI Test BL-34

This experiment represented a 6% cold leg break counterpart test to compare with similar BETHSY, LSTF and SPES experiments. The test was defined to reproduce a range of safety-relevant phenomena in all facilities, the most important relating to the three distinct dryouts and rewets. The first dryout, associated with loop seal formation and core liquid level depression, was followed by a rewet soon after loop seal clearout. Continued mass depletion from the break led to a second core uncovery and dryout, quenched by ECCS accumulator injection. After the accumulator had emptied, further core uncovery caused a third dryout with core temperatures rising sharply to 500°C. This rise was terminated by the low-pressure injection system which produced a sudden rewet which was sustained until the primary pressure was below 7 bar. The test provided transient data for checking code heat transfer, dryout and rewet capabilities at high and low pressures under diverse reactor-typical conditions.

This test has been calculated with CATHARE2 Ver.1.2e/2 Rev. 5 and Rev. 4 using the standard LOBI nodalisation. A full description of the test and relevant calculations being reported in [21].

Predicted results were found to be in good agreement with the experiment for the main observed phenomena such as primary system pressure, break mass flow and mass inventory. Some code difficulties were experienced during the accumulator injection period giving a too early prediction of the LPIS activation. A good liquid level behaviour in the core during the three dryout periods was predicted (***Fig. 1.78***), as was the heater rod temperature response (***Fig.1.79***), although the code tended to overestimate

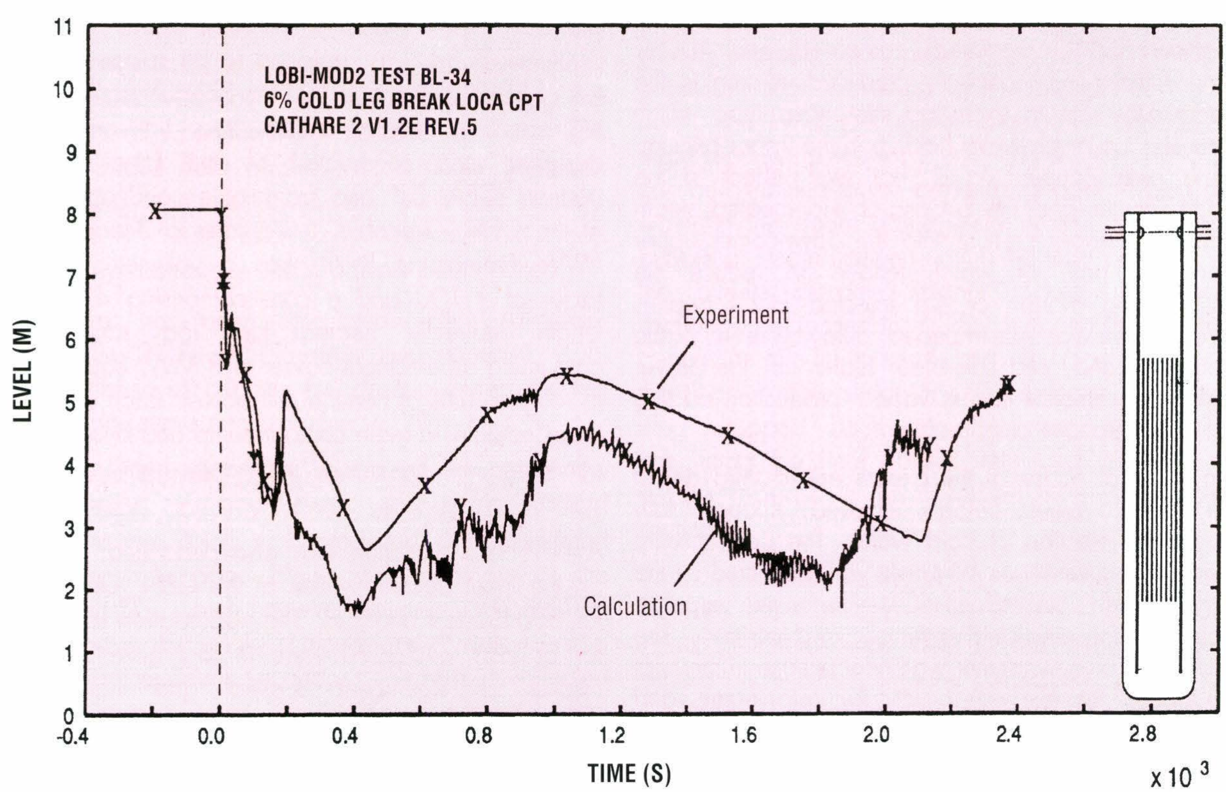

Fig. 1.78 Measured and predicted collapsed liquid levels in core region

1.1.4 EAC-2 - SUPPORT FOR EUREF

In the later part of 1991, a task force of the European Fast Reactor project concerned with whole-core accident calculations, requested a final version of the European Accident Code 2 (EAC-2) code and documentation of the modules which they considered for inclusion into the European Reference Code (EUREF).

For the completion of the combined thermo-mechanics and thermo-hydraulics models of EAC-2 several steps were necessary. First the latest version of the TRANSURANUS pin behaviour code from TUI Karlsruhe was included. This made it again possible to run whole-core cases besides CABRI experiment cases. The code was then cleaned up by running it with the Salford debugging compiler on a fast PC. Then the code was transferred to a HP 700 UNIX workstation because the AMDAHL mainframe computer became unavailable.

As sample cases for the final release, two 10-channel whole-core reactor cases (simulating loss-of-flow and a reactivity accidents) and two single channel reactor type cases were run. The CABRI experiment cases did not run properly on the UNIX workstation. During the transition from steady state to the transient, the special coupling of the CFEM boiling model to the pin behaviour model for CABRI applications led to a non-convergency. When later on the same CABRI cases were run on the IBM mainframe computer at KfK Karlsruhe, these convergence problems did not occur. This indicates that there is a portability problem for this type of application of the EAC-2 code which could not be resolved before the end of the project.

The EAC-2 code system was released to CEA Cadarache, AEA Risley and KfK Cadarache, together with the input and output of the 4 sample cases. These sample cases show clearly one of the main achievements of the EAC-2 development, which is the coupling of a detailed pin behaviour code to a whole-core accident code system and in particular to its in-pin and channel fuel motion models.

As documentation of EAC-2 a new input description and an output description of the MDYN and CAMDYN models were provided. A full documentation of the TRANSURANUS code was also completed. Limited descriptions of the EAC-2 code structure, the CFEM/TRANSURANUS coupling and the main new features of the MDYN model were also provided.

The EAC-2 neutronics package had not been fully coupled to the other parts of EAC-2. Work on certain features of the neutronics package is still underway. This includes the coupling of the quasi static (time-dependent) HEXNODYN code with the static HXSPRT code which is coupled to a cross section generator and has a reactivity worth curve calculation capability. Moreover work on worth curves for transport theory, improved cross section generation, and an improvement of the transport theory itself is still continuing (contracts and an on-going Ph. D. thesis).

This time-dependent 3D transport code is not only of interest to France, the only EC country left to work on Fast Reactors, but also to countries outside the EC. Therefore it was decided to complete the EAC-2 neutronics with the assistance of external experts.

1.2 MANAGEMENT AND STORAGE OF RADIOACTIVE WASTES

An important open-end problem pending solution is the final destination of the "ashes" from nuclear energy generation. A viable technological concept is available with the disposal of the properly conditioned radioactive material into deep geological repositories. However, the implementation of such repositories has nowhere been brought beyond a rather preliminary pilot plant stage.

An inevitable technical consequence is the need to extend indefinitely the storage of conditioned waste and untreated spent fuel in engineered surface storage facilities. In the meantime the idea of a fuel cycle concept, that entails, as a long-term objective, the possibility of individually processed waste components like actinides for a more appropriate immobilisation or complete recycling, has received during the last few years a strong support in the USA, Japan and Europe. Japan has launched a long-term project under the acronym "OMEGA".

On a specific request of the European Parliament, the topic "Potentialities of transmutation of long-lived radionuclides" has been included in the tasks of the 4th R&D programme on Management and Storage of Radioactive Wastes of the CEC, approved by the Council in December 1989 for the period 1990-94.

End 1991 a law was approved in France to postpone any definitive geological disposal for 15 years during which period research on separation and transmutation of long-lived radioisotopes should be performed with the aim to minimise the inventory of long-lived radioactivity in the waste to be disposed of. The timescale and the vastness of research work to be performed is considered in France such to require an international collaboration.

In this scenario a central role of verifying and validating suitable separation and specific conditioning processes can be played by PETRA, i.e. the Plant for Evaluation and Testing of Radwaste management Alternatives, installed in three of the fully licensed ADECO hot cells located in the Nuclear Island of JRC-Ispra, which has been in active operation for more than 20 years.

1.2.1 THE PETRA FACILITY

PETRA is a hot cell facility designed to produce various types of fully active waste such as they will arise from any kind of present and future spent fuel processing with high and ever increasing burn-up. The research programme places specific emphasis on processes suitable to minimise waste (and to improve the quality of the conditioned waste product). In anticipation of the future application of the plant, and besides in its utilisation for potential innovative long-term research, the facility is also seen as a tool for customers and central bodies associated with the quality and characteristics of the final product for performing independent verification and control. An increasing interest has also been shown by Safeguards Authorities for testing under realistic conditions instrumental and analytical methods and procedures for fissile material flow control in processes and waste streams at reprocessing facilities.

Activities

The aim of the PETRA programme during the year has been focused on "nuclear tests" as agreed with the Italian Licensing Authority (ENEA-DISP).

In addition, to achieve adequate plant performance, a series of pre-operational "cold" tests were performed prior to the nuclear tests. The main activities carried out in the PETRA facility can be summarised as follows:

- Several tests were performed using the denitrator to verify both the denitration and Oxal processes. Non-active fission products and "Rare Earths" (simulating the corresponding radioactive isotopes), have been introduced and tested in different chemical flow sheets. Valuable data for future "hot" operation have been collected.

- The fuel chopping machine has been installed inside cell 4411. Subsequent remote handling tests have been carried out to check the complete phase of "feed preparation", i.e. the phase from fuel chopping in cell 4411 up to "hulls" collection after dissolution in cell 4306. Fuel rods have been simulated by means of stainless steel tubes filled with concrete.
- Remote handling operations on liquid and aerosol filters have also been tested.
- Testing of the main processes namely the fuel dissolution process, the first extraction cycle (uranium - fission products separation) and the uranium solidification, have been repeated in order to verify the complete process sequence prior fully active operation. Testing of all other intermediate steps, namely liquid transfer, feed concentration by evaporation, sampling, chemical analyses, etc., have also been repeated. Corrective actions derived from test results have been immediately applied to the plant systems. Operating personnel organisation for running "hot tests" has also been verified during this phase.

The first "nuclear test" concerning the verification of the cell shielding homogeneity has been carried out. Details and results are presented below:

- Two ^{60}Co sources ($1,67.10^{12}$ Bq and $6.66.10^{13}$ Bq respectively) have been utilised for the shielding tests performed by PETRA personnel with the assistance of the Radioprotection Unit.
- A detailed map of exposure rate outside the cell walls has been prepared for PETRA cells (4305, 4306, 4307), and the analytical box (4303). Also the area of the new plug, recently installed in the floor of the cell 4304 to accommodate crucible storage, has been submitted to a shielding test.
- Exposure rate measurements showed good accordance with the shielding design data. Minor nonconformities were detected and the application of necessary corrective measures have been completed during 1992. Measurements were repeated and satisfactory results have been obtained.
- On the basis of the shielding test results we can guarantee for PETRA experiments average exposure rates well below 0.5 μSv/h in all working zones around the hot cells and the analytical box.

The second "nuclear test", started in May 1992, was concerned with three processes, namely the fuel dissolution, the first extraction cycle and the uranium solidification. Non-irradiated uranium and chemical additives simulating the fission products content have been utilised in this test.

- Fuel rods simulating the spent UO_2 fuel rods have been assembled by PETRA personnel using stainless steel tubes as cladding and non irradiated UO_2 pellets (about 10 mm diameter and 0,32% ^{235}U enrichment) as fuel.
- Fuel chopping has been carried out in cell 4411 and a basket containing about 8.8 kg of chopped fuel (6.8 kg of UO_2) has been transferred by remote handling operations through the cells into the dissolver (cell 4306). The above basket content represented the fuel batch for the whole process.
- The dissolution of the fuel batch has been performed by nitric acid at 100°C, followed by cooling, filtration and storage of the feed solution. Plant operating data recorded by the Process Computerised System and chemical analyses performed in cooperation with the analytical staff of the Environment Institute have demonstrated good efficiency of the plant hardware as well as the achievement of a complete UO_2 dissolution. ***Fig. 2.1*** shows a photographic view of the HI601 dissolver and its auxiliary components.
- At the start of the 1st extraction cycle some problems were encountered with the mixer-settlers. The malfunction was due to the high density of the feed solution (never tested before) and the sensitivity of the interphase level control system to the underpressure caused by the off-gas system. ***Fig. 2.2*** shows a photographic view of the complete bank of mixer settler batteries located in cell 4307. Discrepancies between expected and experimentally determined profiles of mixer settlers stages were made evident by an adequate programme of chemical analysis. After a few minor piping modifications, very satisfactory results have been achieved. The final "product", ie. fission product free uranyl nitrate, has been concentrated (approximately 280 g/dm^3) in the denitrator and then reutilised as the "feed solution" for a second test run. Secondary processes such as mixer settler batteries rinsing and organic solvent clean-up have also been successfully tested.

After process completion two main liquid streams were collected namely:

- A uranyl nitrate solution, ready for the subsequent uranium solidification process.
- An aqueous solution containing simulated fission products, ready for the subsequent denitration - oxalate precipitation and vitrification processes.

On the basis of results of "nuclear tests" we can conclude that the five PETRA plant safety systems have successfully met the imposed requirements. No defects have been detected during the entire series of nuclear test runs. The organisation which was set up for the test programme has achieved the proposed objectives. The Process Computerised System, telemanipulators and glove boxes have sussessfully been utilised by the operator team.

During 1992, under the auspices of the JRC-ENEA collaboration a mixed team of JRC-ENEA operators was created and successfully employed for PETRA operation.

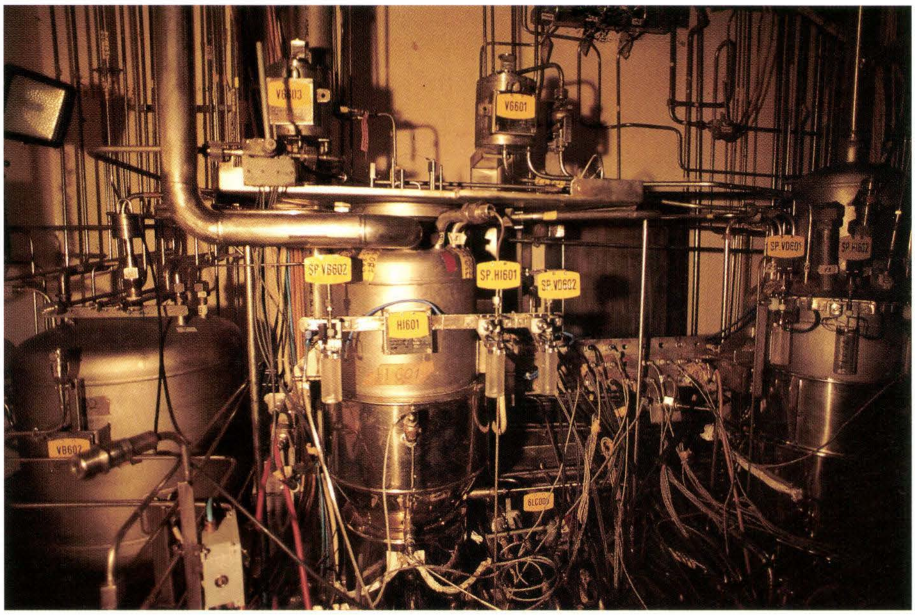

Fig. 2.1 View of dissolver assembly (cell 4306)

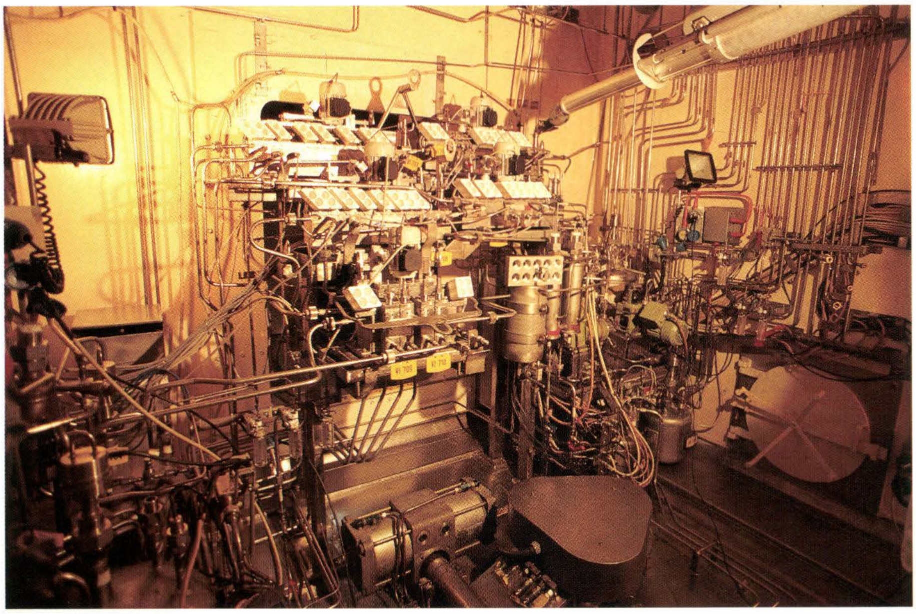

Fig. 2.2 View of mixer settler batteries (cell 4307)

Licensing

Main achievements during 1992:

- Safety Authority approval of the "Nuclear Test Programme" (January).
- Safety Authority authorisation of the 1st nuclear test (March).
- Inspections during the 1st nuclear test and its repetition after corrective actions (May-June).
- Certification of positive nuclear test results and subsequent authorisation of the 2nd nuclear test (June-July).
- Inspection during the 2nd nuclear test (September-December).

Planning 1993

The current planning foresees the following steps:

- Completion of the second nuclear test.
  The uranium solidification process will be carried out in January.
  Aqueous wastes will be further treated (end of February) in order to recover all uranium residues in the plant vessels.
- At the time being, a decision is pending on the future activity in PETRA. Two options are now being considered for PETRA utilisation:
  a) Treatment of Highly Enriched Uranium (HEU) fuels performed in the frame of a Service Contract.
  b) Experimental exercises using non-irradiated uranium and simulated non-radioactive fission products.
- Option a) will require a deep technological assessment of the PETRA plant performances entailing:
  - A new safety assessment concerning criticallity conditions during the fuel processing and product storage.
  - The set-up of new chemical flow sheets designed for processing low burn-up HEU fuels.

  Modification of some plant hardware and setting up of analytical support facilities will also be considered.
- Option b) will include:
  - the execution of denitration- oxalate precipitation, 2nd extraction cycle, vitrification of simulated fission products (i.e. the separated chemical additives),
  - exercises on remote handling operations,
  - the implementation of plant modifications in compliance with test results, concerning in particular:
    – a new fuel transportation device
    – the instrumentation upgrading
    – the remote manipulation tools
  - the execution of a new complete test run using unirradiated uranium rods.

1.2.2 MONITORING OF RADIOACTIVE WASTE

Before storing radioactive waste, its content in radioactive nuclides emitting alpha, beta and gamma radiation has to be determined. Waste containing fissile (uranium and plutonium) and non-fissile (gamma emitting nuclides) contaminants has to be characterised. In contrast with safeguards requirements, two factors make a precise measurement difficult and the application of special techniques mandatory. The two factors are:

- the presence of a mostly unknown and inhomogeneous waste matrix and
- the random location of the radioactive contaminants in the bulk of this matrix.

The research activity in the field of waste monitoring has been developed and applied according to:

- active neutron interrogation methods for determining uranium content
- passive neutron interrogation methods for determining plutonium content
- gamma interrogation methods for characterising gamma-active waste.

Measurement of uranium waste by active neutron interrogation

Non-destructive neutron interrogation of uranium is effective only if active neutron interrogation is applied. The existing active neutron interrogation devices are: active well coincidence counter, Cf-shuffler, differential die-away and PHONID [1-3]. The latter one has been developed at the JRC Ispra and has shown its capabilities to measure homogeneous bulk material samples for safeguards purposes [1,2] with a precision of the order of 2%. However, for waste materials the accuracy of all mentioned devices does not reach the diagnostic performance level imposed by the Safety Authorities for disposal. This is due to the fact that in principle these devices measure only one signal from the induced fission neutrons. This signal is not enough to yield information about the moderating power of the sample, the attenuation of the interrogating neutron beam, the efficiency etc. Even for bulk material all those devices show their diagnostic shortcomings especially in case of moderating matrix materials. To overcome this problem a research programme has been set up at the STI of JRC Ispra.

In a first step a measuring campaign, in collaboration with BNFL Springfields, has been performed to define the main shortcomings of a current active neutron interrogation device [4]. On the basis of the results of these measurements a proposal has been made for the development of a new active neutron interrogation device which should be capable of measuring U-waste with the accuracy required. The development of such a system was based on the MCNP* code calculations in parallel with measurements.

In October 1992 a modified version of the PHONID 3b has been used to validate one of the basic assumptions of the proposal. A special ^{3}He detection bench, consisting of normal ^{3}He detectors and ^{3}He detectors surrounded with a layer of Cd, has been installed in the existing device. Such an arrangement was aimed at achieving additional information of the matrix based on transmission, measurements of the Sb-Be source neutrons. In addition to the measurements performed at BNFL, laboratory measurements have been set-up to check the possibiity of using position sensitive ^{3}He neutron detectors for neutron transmission measurements. The transmission of neutrons through U-Al plates was studied. The characteristics of the U-Al plates are reported in *Table 2.1*.

The neutron source used consisted in an Am-Li (α,n) source encapsulated in a polyethylene cylinder with

Table 2.1 Characteristics of the U-Al plates used in the neutron transmission measurements

U-weight	16.115 g
Enrichment	92.3%
Dimensions	58.3 x 103.0 x 3.15 mm^3

* Montecarlo Neutron Photon transport code

a diameter of 100 mm and a height of 100 mm. In *Fig. 2.3* the position spectrum of the neutron source without any absorbing material between the source and the position sensitive neutron detector is compared with that of a neutron source with six identical U-Al plates between the source and the detector.The width of the valley in the second spectrum corresponds to the dimension of the U-Al plates. The difference of both spectra in the region where the neutrons are absorbed (between channel numbers 240 and 330) is fitted by a Gaussian plot. The absorbed neutron flux is plotted (*Fig. 2.4*) versus the number of U-Al plates between the source and the detector. The curve in *Fig. 2.2* is the result of a weighted least square fit procedure based on the relation:

$$C = a\,(1 - e^{-bn})$$

where **a** reflects the neutron flux of the source and **n** is the number of U-Al plates. The fitting constant **b** (0.42±0.02) is related to the absorption cross section of the material in question. Using a surface density of 0.3g cm^{-2} of absorbing material the constant **b** results in a microscopic absorption cross section of 185 b.

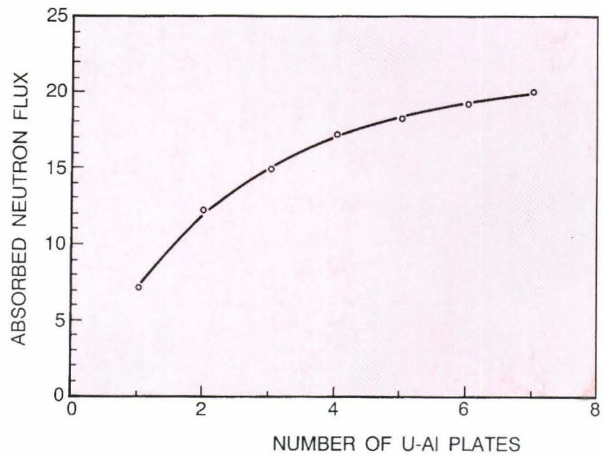

Fig. 2.4 Neutron flux absorption

Measurement of plutonium waste by passive neutron interrogation

The activity in the field of Pu waste monitoring was subdivided in the following main actions:

- Field tests with waste barrels in a MOX fuel fabrication plant
- Design of a new industrial detector head
- Theoretical treatment of the signal pulse train
- Technical advices to the contractor A.N. Technology Ltd. for the TCA technique
- Collaboration with the Research Centre of Casaccia (ENEA, Italy)

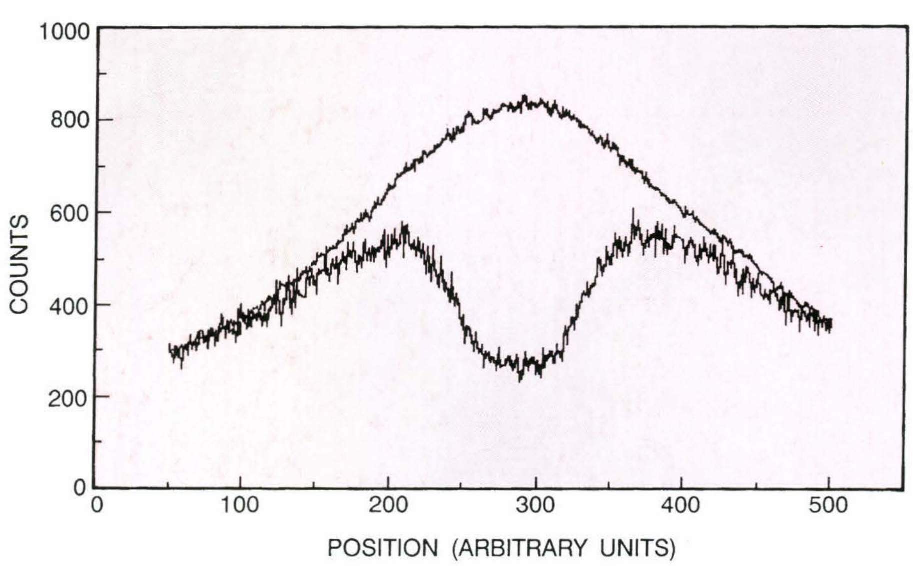

Fig. 2.3 Position of sensitive neutron detector

Field tests

The neutron signal correlation technique was used to measure the ^{240}Pu mass equivalent inside 220 dm^3 waste barrels. For this purpose a 4π-neutron detector head, with 60 ^3He neutron detectors of 100 cm length, was used. The resulting signal pulse train was generated by means of two different signal frequency analysers. The first one manufactured by the Rogowski Institut (Aachen) permitted the measurement of the signal frequency distribution of 16 different observation interval sizes τ_i for signal and periodically triggered intervals. The second analyser was manufactured by ASEM

(Udine) and delivered only the frequency distribution for signal triggered observation intervals. The purpose of the measurements was to test the performances of pair and triple correlations on real waste of a fuel fabrication plant in comparison with the electronic equipment and the used interpretation models. This waste was composed of 15 to 35 small waste packages loaded into a 220 dm^3 standard waste barrel. Each small waste package was measured before in a dual range detector head using the correlated signal pairs R_2 for the ^{240}Pu mass determination. The α-ratio of the waste was in the range of 1 to 10.

For determining the ^{240}Pu mass equivalent in the Pu contaminated waste of a 220 dm^3 barrel, three different interpretation procedures were used:

- Calibration via the correlated signal pair rate,
- Absolute measurement using the declaration of the operator for the isotopic composition (known as α-ratio),
- Absolute measurement without data of the operator.

All three methods were applied using the measurement data of the two signal correlation techniques employing signal triggered and periodically triggered observation intervals. The best results were obtained by the absolute method, without using any operator data.

The ^{240}Pu mass equivalent, m(Pu240), was measured with a precision of about 6.5% for an α-ratio below 3. Large errors in the ^{240}Pu mass equivalent measurement were found for barrels with an α-ratio of about 10 (*Table 2.2*). This high α-ratio was detected for a MOX plant waste although the operator declared value was below 1. A measurement time longer than 30 min for barrels with α-ratios above 3 would improve the assay result. More details of these measurements are given in ref. [5].

Table 2.2 Comparison of barrel measurement results with declared m(Pu240) values

	Barrel Code N°	Declared m(Pu240)	Measured	
			m(Pu240)	α-ratio
		[g]	[g]	
1	ZL38161	5.89	6.31±0.23	1.01±0.40
2	ZL38121	5.89	6.52±0.46	1.59±0.09
3	ZL38171	5.89	5.27±0.25	1.90±0.06
4	ZL3882	6.04	5.59±0.80	1.92±0.23
5	ZL38071	5.59	5.64±2.70	2.04±0.05
6	ZL38181	5.89	5.39±0.63	2.22±0.19
7	ZL38191	6.12	6.64±2.70	2.48±0.80
8	ZL38151	5.89	5.24±0.27	2.51±0.10
9	ZL3869	5.91	6.93±0.93	4.28±0.35
10	ZL3870	5.76	6.19±2.43	5.87±1.27
11	ZL3877	6.39	8.30±6.70	10.10±4.28
	Sum 1-11	65.26	68.02±7.76	α > 1.00
	Sum 1-8	47.20	46.60±2.96	α > 3.00

Design of a new industrial detector head

A new 4π-neutron detector head was designed, based on the latest safety requirements for nuclear plant equipment. The detector head was designed to have steel or Al cladded polyethylene moderators and neutron shielding. An upright handling of the waste barrels was foreseen, with barrels manipulated by a crane and inserted into the cavity, which has the form of a decagon, via two horizontally opened doors.

A moderator to ^{3}He neutron detector volume ratio of about 10 was chosen for the neutron detector modules to provide a high neutron detection probability and a light undermoderation of the neutrons. The neutron detection probability was close to 20% with a decay time of about 70 μs. The modules can easily be rearranged for being operated either in the under or overmoderated range. To achieve an even higher detection probability for waste measurements on barrels having similar matrices (concrete or bitumen), the Cd plates of the modules facing the cavity can be removed. In such conditions the correlation analysis has to be performed with only one observation interval of 2 or 3 times the decay time of the detector/waste barrel arrangement.

Theoretical treatment of the signal pulse train

For the analytical treatment of the factorial moments of the probability distribution a set of formulae was derived, which permits a correction of the paralysing neutron dead time. Formulae of this kind were published in ref. [6] for signal triggered observation intervals. Similar formulae are in elaboration for periodically triggered observation intervals and will be summarized in ref. [7].

For a more general treatment of the neutron correlation technique a multivariate probability generating function was derived [8]. With this function the probability distribution and its factorial moments can be expressed as function of correlated multiplets. The use of Bell polynomials gives for both quantities explicit expressions [9,10]. This is valid for signal and periodically triggered observation intervals. It is believed that the probabilities are much less dependent on the neutron multiplication as the factorial moments. For this reason it is extremely useful to derive analytical expressions for the ^{240}Pu mass assay based on a measurement of these probabilities.

Technical advices to Antech (AN) Technology, Ascot, UK

AN has concluded a contract with the STI for a transfer of the Time Correlation Analyser (TCA) technique know-how existing at the JRC Ispra. This technique has been applied successfully in the past to measure Pu in form of bulk material or dispersed in radioactive waste with low α-contamination.

In the frame of this contract AN was provided with the following sources of information concerning this technique:

- A functional description of the instrument
- Two technical proposals for a TCA, one based on a shift register, the second on gates and FIFOs
- A computer code for data acquisition, elaboration and interpretation valid for the JRC Ispra TCA instruments and the corresponding FORTRAN source code. This software included Pair and Triple Correlation routines for two different signal correlation methods, all applicable to three different measurement tasks: F.E.S.A., F.E.M., SA.F.M. (unknowns: fission rate F., detection probability E., (α,n)-neutron emission rate SA. and neutron multiplication M.).

Several meetings were held at the JRC Ispra involving neutron physicists and electronic engineers to give advice for the redesign of the TCA hardware based on latest chip developments.

Collaboration with ENEA, Casaccia (Italy)

Advice was given to ENEA Research Centre of Casaccia for the installation of a measurement laboratory foreseen for the assay of Pu contaminated waste. The matter of collaboration concerned the design of a 220 dm^3 4π neutron detector head and the interpretation of measurement data.

Characterisation of gamma active waste

During the decommissioning and operation of "hot" installations (e.g. "hot" cells) waste contaminated with fission products, mainly Cs and Co, are produced. Before storing and conditioning the 220 dm3 compressed waste drums with a matrix of hydrogeneous materials (gloves, paper, etc.), have to be measured to determine the most important gamma emitters present.

The boundary conditions are:

- fast measurement execution,
- determination of whether or not a prefixed level of contamination is exceeded,
- determination of individual contaminant nuclides.

A simple system has been set up based on a Ge detector without collimator, detecting the activity of the whole drum. The intrinsic efficiency of the Ge detector was determined by standard sources, for a distance detector-drum centre of 42 cm.

The waste matrix had a diameter of 38 cm and a height of 76 cm. The intrinsic efficiency for sources placed on different radii in the detector-drum centre plane did not change by more then 3% if the drum was rotated arount its axis. For sources placed up to 70 cm above the drum centre on the axis of the drum the deviation from the centre intrinsic efficiency was not more than 10%. The greatest deviation of about 15% occurred when the sources were placed at the most outside position of the waste matrix, at the top or bottom and at a radius of 19 cm. For a distance detector/drum centre larger than 42 cm the intrinsic efficiency becomes more homogeneous.

As the intrinsic efficiency does not vary much in function of the position of a source in the waste drum, does not vary much, the detector was calibrated with known gamma emitting sources placed in the centre of an empty drum. The standard deviation between the fitted and measured efficiencies for the 7 source energies was 1.9% for sources with a declared uncertainty of 2.5% and a statistical counting uncertainty of 0.6%.

In the matrix material the gamma rays are absorbed or lose energy by Compton scattering. For a typical non-contaminated matrix the total absorption coefficient and the matrix homogeneity have been determined. Gamma sources were placed in the centre of the rotating drum as well as in a position outside the rotating drum. Results are reported in ***Table 2.3*** where the average absorption value, currently expressed in $cm^2\ g^{-1}$, is a relative value to the value of the iron absorption [11]. Indeed, it results that within experimental uncertainties the shape of the matrix absorption follows that of iron. The uncertainty of the absorption relative to iron was determined, taking account of the absorption calculated for the different gamma peaks for one single path or measurement, it amounts to about 1% standard deviation.

By utilizing 4 of the 5 paths, as can practically be done, the true matrix homogeneity was normally underestimated. Only for measurement 1 in which the path crossed the drum in different directions, a

Table 2.3 Gamma-ray absorption of a typical matrix

Meas. no.	Gamma sources	Source position	Average absorption relative to iron	No. of paths per drum	Inhomogeneity of drum (%) (one standard deviation)
1.0	^{137}Cs, ^{60}Co	centre	0.958±0.022	16	11.8
2.1	^{137}Cs, ^{60}Co, ^{106}Ru, ^{88}Y, ^{65}Zn, ^{22}Na	outside	0.949±0.039	1	–
2.2	^{137}Cs, ^{60}Co	outside	0.992±0.012	5	5.3
3.1	^{137}Cs, ^{60}Co	outside	0.961±0.009	5	2.4
3.2	^{137}Cs, ^{60}Co, ^{152}Eu	outside	0.937±0.015	1	–
4.1	^{137}Cs, ^{60}Co	inside	0.936±0.029	1	–
4.2	^{137}Cs, ^{60}Co	outside	0.934±0.010	1	–
5.0	^{152}Eu	outside	0.949±0.015	5	1.1

reasonable estimate of the homogeneity was possible. However, the average absorption of the matrix was determined with an acceptable precision for all the measurements.

The largest source of uncertainty is due to the unknown location of nuclide contaminants in the drum. This uncertainty was about 30% for the two extreme cases, i.e. contaminants in the centre of the drum or on the border of the drum relative to a supposed homogeneous distribution. In order to eliminate this source of uncertainty the use of a sophisticated system with collimator and tomographic techniques was necessary leading to long measurement times.

Due to the fact that in a real waste drum the source position is unknown, an analysis method was chosen in which the maximum content (i.e. the "UPPER BOUND") of the gamma contaminants is determined. It was decided to divide the drum in five horizontal sectors of equal heights measuring the gamma spectrum with the detector in front of each sector of the rotating drum. Assuming five sources in the centre of each sector the upper bound of the contaminant activity in the drum was determined.

Test measurements were executed using couples of calibrated ^{137}Cs and ^{60}Co sources. These couples were placed in the mid and top plane positions of the drum central axis or of the matrix border. Results are reported in ***Table 2.4***. For all source locations in the drum central axis position, values close to the nominal source activity were measured, whereas for all locations in the matrix border position, the measured values were about twice the nominal source activities. The small difference between the nominal values and those measured in the central axis was due to the non-homogeneity of the matrix absorption.

On a laboratory scale 10 real waste drums have been characterised by measuring the absorption of the matrix by neutron transmission with an Europium source. The inhomogeneity (as measured with 5 paths) is of the order of 20%. The upper bound of the amount of contaminants was due mainly to ^{137}Cs and ^{60}Co, whereas minor quantities (less than 1%) were due to ^{134}Cs and ^{154}Eu.

The average absorption value relative to iron determined for the 10 drums was:

$$\Sigma = 0.96 \pm 0.2$$

and in agreement with the absorption for a typical matrix (***Table 2.3***). In ***Table 2.5*** the activity values measured for a single typical waste drum are shown.

In agreement with ENEA-DISP, using a single identical absorption value for all drums (e.g. $\Sigma=1.0$), each single drum was measured with a certain error. However, if a total activity value is measured by cumulating a certain number of drum measurements, the average activity of a single drum can be

Table 2.4 Test measurements using calibrated sources

source	nominal source activity (kBq)	Measured source activity (kBq)					
		central axis position			matrix border position		
		mid plane	top plane	mid & top planes	mid plane	top plane	middle & top planes
^{137}Cs	432	402	423	412	926	932	929
^{60}Co	317	316	339	327	567	579	573

Measured source activity (kBq)					
source	minimum	maximum	average	maximum deviation	standard deviation
^{137}Cs	402	932	667	40%	23%
^{60}Co	316	579	448	29%	17%

Table 2.5 Results of gamma-activity measurements performed on a single waste drum

barrel		**measured nuclide**	**activity (MBq)**	**distribution (%)**
- Code n°	178	$137C_s$	5.2035E+01	91.02
- Volume (dm^3)	210			
- Weight (kg)	110			
waste		60Co	4.7954E+00	8.39
- tot. Act. (MBq)	5.7168E+01	$134C_s$	0.1703E+00	0.30
- origin	ENEA	^{154}Eu	0.1675E+00	0.29
- nature	(paper, plastic, cotton, rubber, glass, etc.)			

correctly determined since the positive and negative single drum errors compensates each other. The instrument has been installed at the Radioactive Waste Conditioning Building of the Ispra Site and is now used on a routine basis.

References

[1] SPRINKLE J.K.,BARDELLI R., BECKER L., LEZZOLI L., ROCHEZ R., SCHILLEBEECKX P., WENG U. - The measurement capabilities of PHONID3b - Proceedings of the 13th ESARDA Symposium, Avignon (F), 14-16 May 1991

[2] SPRINKLE J.K., BARDELLI R., BECKER L., LEZZOLI L., ROCHEZ R., SCHILLEBEEKX P., WENG U. - The capabilities of PHONID 3b - 4th Intern.Conference on Safeguards, ANS 29, Albuquerque, NM (USA), Sept 4 Oct 1991, Proceedings Capture gamma-ray spectroscopy, Pacific Grove, Ca 199O AIP, Conference Proceedings n.238 (1991), p. 257

[3] BONDAR L., D'ADAMO D., DIERCKX R., HAGE W., HUNT B., PEDERSEN B., SCHILLEBEECKX P., VOCINO V. - Measurement facilities for radioactive waste at the JRC Ispra establishment - Proceedings of the 14th annual ESARDA meeting, Salamanca (E), 5-8 May 1992

[4] NAPIER S., SCHILLEBEECKX P. - PHONID 3b measurements of uranium waste - EUR 14551 EN (1992)

[5] BONDAR L., HAGE W., PEDERSEN B., SWINHOE M., HAAS E., LEDEBRINK F.W., AMELING W., KLEINEKORTE K., KARAVAS A. - Pu waste measurement in a MOX fuel fabrication plant by neutron correlation methods, Technical Note N° 1.92.99, (1992)

[6] HAGE W., CIFARELLI C.D.M. - Correlation analysis with neutron count distribution for a paralyzing dead-time counter for the assay of spontaneous fissioning material - Nucl. Sci. and Eng. 112, pp. 136-158, (1992)

[7] HAGE W., PEDERSEN B. - Summary of formulae for the dead time correction of signal doublets and triplets in various types of observation intervals - EUR report in preparation

[8] HAGE M., CIFARELLI C.D.M. - A multivariate characteristic function for signal sequences - EUR report, Commission of the European Communities, to be published

[9] HAGE W., PEDERSEN B., BONDAR L., SWINHOE M., LEDEBRINK F.W. - Triple neutron correlation for MOX waste assay - ANS 1993 Annual meeting & ANS topical meeting on National critical technologies, Advanced technology applications, San Diego, Ca, USA, June 20/24, 1993

[10] BONDAR L., D'ADAMO D., DIERCKX R., HAGE W., HUNT B., PEDERSEN B., SCHILLEBEECKX P., VOCINO V. - Measurement facilities for radioactive waste at the JRC Ispra establishment - 14th Annual ESARDA meeting, Salamanca, Spain, 5/8 May, 1992

[11] Photon Cross Section Attenuation Coefficients, and Energy Absorption Coefficients. NSRDS-NBS 29 (August 1969)

1.3 SAFEGUARDS AND FISSILE MATERIAL MANAGEMENT

At the JRC a specific programme is being developed which is intended to provide improved technical tools suitable for verification activities. The basic justification for this R&D activity performed within the Commission's Frame Work Programme is laid down in its legal obligations:

- *Safeguards on nuclear materials is performed in nuclear installations within the European Community (EC) in the framework of the Euratom Treaty (Chapter VII, articles 77-85) and Supply Agreements. The control activities are carried out by inspectors from the Euratom Safeguards Directorate (ESD, DGXVII Luxembourg).*
- *In the frame of the Non-Proliferation Treaty (NPT) and the Verification Agreements, the International Atomic Energy Agency (IAEA) is performing verification activities worldwide, including the EC.*

Safeguards scientific and technical activities performed in the frame work of the programme R&D for Safeguards and Fissile Material Management illustrated below are tightly coordinated with and complementary to the Support to the Commission activities described in paragraph 2.

The main safeguards activities performed at the Safety Technology Institute (STI) of the JRC Ispra are executed within the Safeguards PERformance LAboratory (PERLA) of the Institute (see Appendix A.2 of ref. [1] and A.4 of this report).

They deal in particular with Non Destructive Analysis (NDA) methods for the determination of U and Pu isotopes in bulk and itemized form by active and passive neutron measurements, γ-spectrometry, calorimetry, used either individually or in integrated systems.

In 1992 a large effort has been dedicated to finalize and start up new facilities: PETRA (Appendix A1 of ref. [1]) and TAME laboratory [1] are now ready to be used for their respective programme, i.e. fuel reprocessing and waste characterization (PETRA) and tank volume measurements (TAME). Both of them will be largely employed for Safeguards R&D and supporting activities.

*With the start-up of operations of PETRA and TAME, a large portion of the back end of the nuclear fuel cycle is represented by STI facilities (**Fig. 3.1**):*

- *PERLA is to a certain extent representative of Low Enriched Uranium (LEU), High Enriched Uranium (HEU), and Mixed Oxide (MOX) fuel fabrication plants*
- *PETRA and TAME are representative of reprocessing facilities.*

*In the field of safeguards the STI nuclear facilities constitute a tool unique in the world and perfectly aligned with the **subsidiarity** mandate of the Commission.*

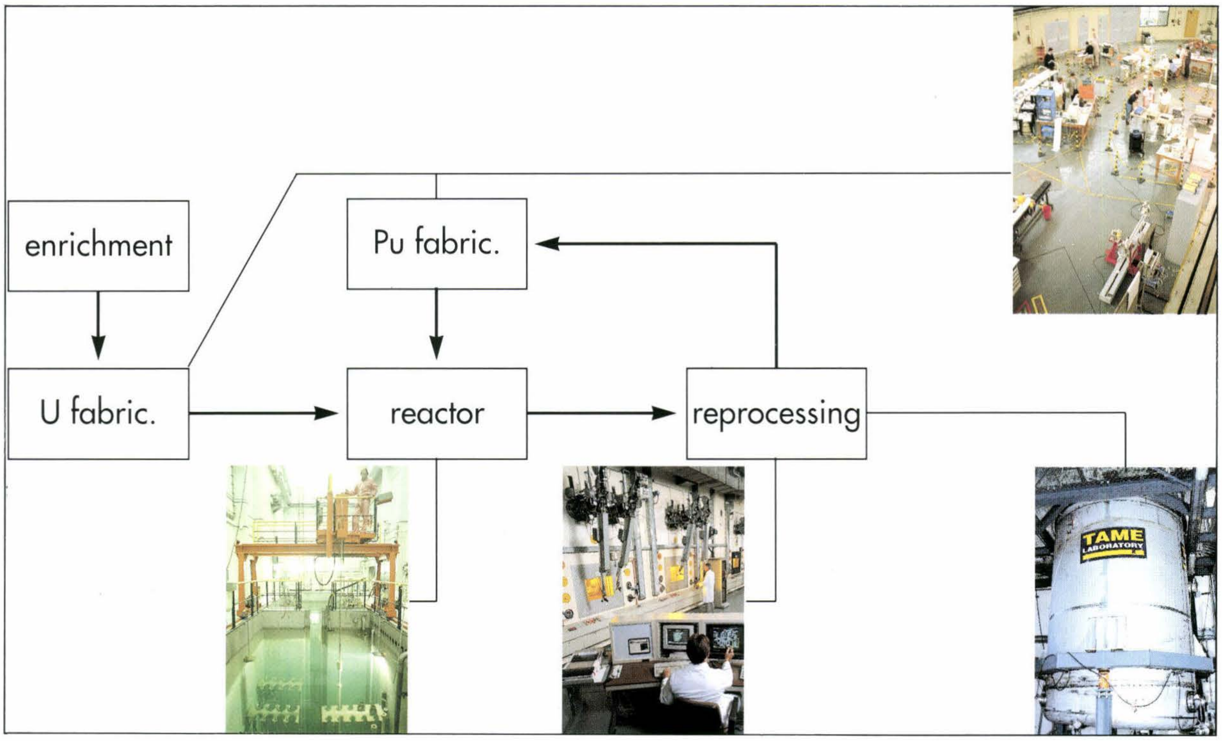

Fig. 3.1 Fuel cycle represented by STI facilities

1.3.1 NON-DESTRUCTIVE ANALYSIS (NDA) METHODS FOR SAFEGUARDS

MTR fuel element monitoring

In the framework of the PERLA activity for the monitoring of High Enriched Uranium (HEU) Materials Testing Reactor (MTR) fuel elements [1], an R&D study was conducted in collaboration with the Università La Sapienza, Dipartimento Energetica, Roma (Italy) to set up a methodology that on the one hand could reduce the need for MTR standard fuel elements and on the other hand could lead to a field data interpretation software, tailored for inspector use of the MTR scanner (*Fig. 3.2*).

The experimental programme considered three steps:

- execution of experiments and data analysis,
- validation of the MERCURE Montecarlo code using the above experimental results,
- creation of a model allowing the evaluation of ^{235}U content in fuel elements, being geometrically and physically different from the standards.

Results showed good agreement between MERCURE and the experimental data, legitimating the use of MERCURE to compare with the model results and validate them. In *Fig. 3.3* a typical comparison is

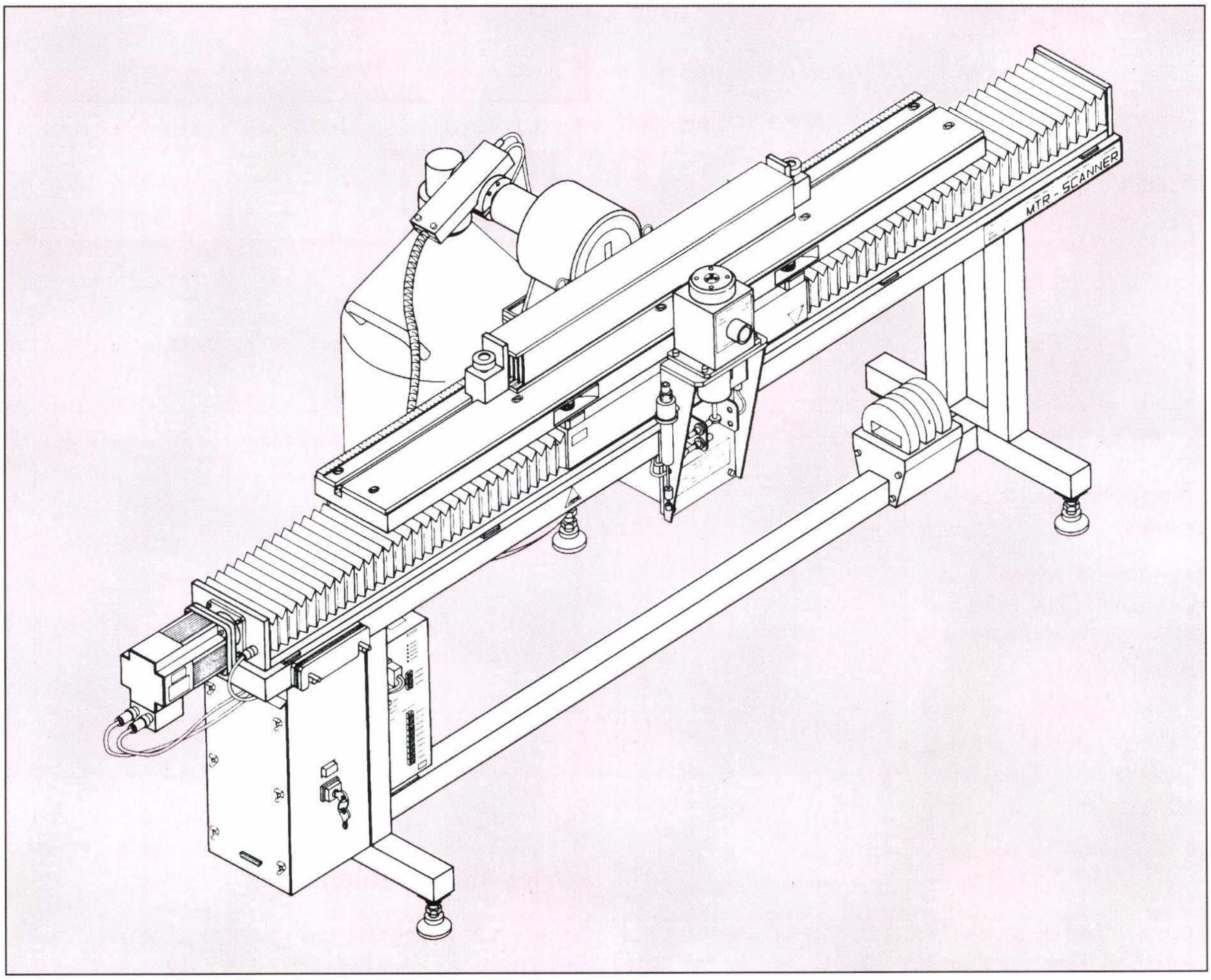

Fig. 3.2 MTR scanner

Finally, another peculiar aspect is that the HRGS random error is also lower for P-eff than for ^{240}Pu-eq, showing that the set of photopeak areas used to determine the latter and their relative propagated weight through the HRGS models, gives rise to a larger counting statistics uncertainty.

Conclusions

The most prominent conclusions reached are the following:

- Calorimetry for plutonium (and tritium) is an accurate technique and sometimes an even more accurate level than Passive Neutron Counting (PNC).
- It is not influenced by most perturbations that normally affect NDA methods.
- Its measuring times are substantially longer than those of PNC techniques especially if compared with Shift Register based devices. Multiplicity counters may in fact require counting times close to calorimetry time in order to attain comparable precisions.
- Finally the calorimetry must not be seen as a competitive or alternative technique to PNC, but as complementary.

Complementarity can be exploited in two ways:

- Inhomogeneous samples with impurities and moisture giving unsatisfactory results and/or significant discrepancies, can be remeasured by the calorimeter.
- Sample strata that are known to be affected by the above perturbing parameters or poorly known (^{242}Pu, (α, n)) can be sampled for regular use of calorimetry and PNC. A few samples could then be measured by both techniques and the data analysed by an integrated model accounting for HRGS+CAL+PNC [4]. Disposing of a more independent equation, the uncertainty linked to ^{242}Pu can be dramatically reduced, giving accuracies very much closer to those of power measurements and indeed very close to those of chemical analysis.

Active measurements of PERLA uranium samples

Measurements of uranium PERLA samples were performed using the active interrogation technique with Am-Li sources. Two detector heads were used, one commercial Active Well Coincidence Counter (AWCC) and the other built at JRC. The Shift Register electronics was used, always driven by the Active Euratom Coincidence Counter (AECC) software running on an IBM compatible personal computer.

The detector heads were used in different configurations, with or without Cd liners in order to cut or not the thermal flux going back to the uranium sample. The configuration with Cd is called "fast mode" and that without Cd "thermal mode". An adaptor with a special insert is used to house the MTR element. Results are given in ***Table 3.1***.

Table 3.1 Active measurements of uranium PERLA samples

Material	Physical form	Enrichment %	Weight ^{235}U(g)	Accuracy %	Detector configuration
UO2	powder	19.8	>300	5	thermal mode
UO2	powder	35	>300	1	thermal mode
UO2	powder	60	>300	2	thermal mode
UO2	powder	92.7	>300	2	thermal mode
UO2	pellets	19.8	>300	2	thermal mode
U	metal	93.1	>1000	1	fast mode
U/Al MTR	plates	19.5	>1g/cm	4	MTR mode
U/Al MTR	plates	93.2	>1g/cm	3	MTR mode

Combined use of neutron correlation technique, calorimetry and gamma spectroscopy for non-destructive analysis of plutonium isotopes [4]

The assay of a plutonium sample has been performed using a High Level Neutron Coincidence Counter (HLNCC) together with a gamma ray isotopic determination. This technique may improve the methods used to estimate the 242Pu content and (α,n)reaction rate. The use of another parameter such as the thermal power of plutonium samples measured with a calorimeter, gives an additional relationship which links the unknown weights of plutonium isotopes and 241Am to the measured quantity.

A method has been developed to evaluate the plutonium and americium isotopes masses, in particular 242Pu, which gives a better estimation of the total plutonium mass. The algorithm uses the measurement results from HRGS, HLNCC and calorimetry.

The method allows also to evaluate the (α,n) reaction rate. In fact, when the plutonium mass is measured with calorimetry combined with data on plutonium isotopic composition and 241Am content obtained by destructive analysis, 240Pu-eq is also calculated. Then the neutron equation with multiplication correction have to be used to evaluate the (α,n) factor, without any correlation.

The proposed technique gives an accurate determination of the plutonium mass, without prior knowledge of the 242Pu content. The samples assayed gave an average absolute error on the plutonium mass of 0.034% and a standard deviation of 0.52%, much better than what can be achieved with HLNCC-MGA (mean=1.72%,S.D.=1.22%) or with CAL+MG (mean=1.62%,S.D.=1.12%). The combined technique is also more accurate in the determination of α and M factors. Therefore it can be used to give an accurate estimation of α for impure plutonium material.

In future will be evaluated the use of this method to get an estimation of the alpha factor (alpha-n reaction rate), of the multiplication factor M and of the 242Pu abundance with samples not so pure as PERLA standards and in other chemical forms.

Plug-in card for data acquisition for pulse to pulse analysis of a neutron pulse train (PIA: Pulse Interval Acquisition)

The system measures the time intervals between successive signals and stores them in the hard disk of a Personal Computer for further analysis. Such a system is used to study the long-term behaviour of the signal background due to cosmic radiation neutron bursts, direct interaction of cosmic radiation with the neutron detector wells and electronic disturbances. The incoming signals of a pulse train with a signal width of 50 µs are syncronized with 5MHz. The time intervals between successive signals are memorized in the memory of the card and then stored in the Hard Disk of a PC for further analysis. The signal resolution is 30 µs [5].

Background measurements were performed with three different instruments: HLNCC, AWCC and TIBO 32. Multiplicities up to 7, 18 and 44 were found for these instruments. Further analysis have to be carried out in order to better investigate these effects and to improve the sensitivity to the small plutonium mass using the correlation technique for the analysis of the pulse train.

It is planned to apply the PIA card to the measurements of radioactive wastes, which means small quantities of plutonium varying from 10 mg to 100 g, as well as to the safeguards measurements, i.e. to quantities of plutonium ranging from 1 g to 2500 g.

Plutonium pin assay (PUPA)

The PUPA counting system assays plutonium content of fuel pins. It counts coincidence neutrons from the spontaneous fission of plutonium and it may accommodate various kinds of pins and rods having diameters less than 20 mm and lengths up to several meters. PUPA is comptabile with other detectors in the HLNCC II family of instruments.

The detector is portable and, if desired, it can fit into a carrying case for transport and operation. The system consists of the PUPA, the Shift Register, and the portable PC with the PECC (Passive Euratom Coincidence Counter) software. PUPA, designed at the JRC Ispra laboratories of STI, is used for ESD inspector training. A laboratory version of PUPA allows to optimize its design to meet specific customer requirements.

The PUPA counting system has been tested and deployed under routine Euratom inspection. The instrument is permanently installed at the VENUS facility of Mol, Belgium.

1.3.3 PROGRESS IN PERLA

The PERformance LAboratory (PERLA) is a complex of laboratories (see Appendix A.2 of ref. [1] and A.4 of this report). One of them, the New-PERLA facility has been completed in 1991 and has started operation in August 1992 (see Appendix A.4 of this report).

Significant experiences such as testing, training and calibration exercises on the SIEMENS Plutonium Input Measuring Station (PIMS) have been conducted in the new laboratory. As reported in paragraph 2.1, the number of training courses held in PERLA during 1992 on Non-Destructive Assay, statistics and safeguards verification techniques has been significantly increased. To cope with the training issues it is the intention of the STI management to reinforce the training structure by setting up the Safeguards PErla Centre for TRAining (SPECTRA), see *Fig. 3.8*.

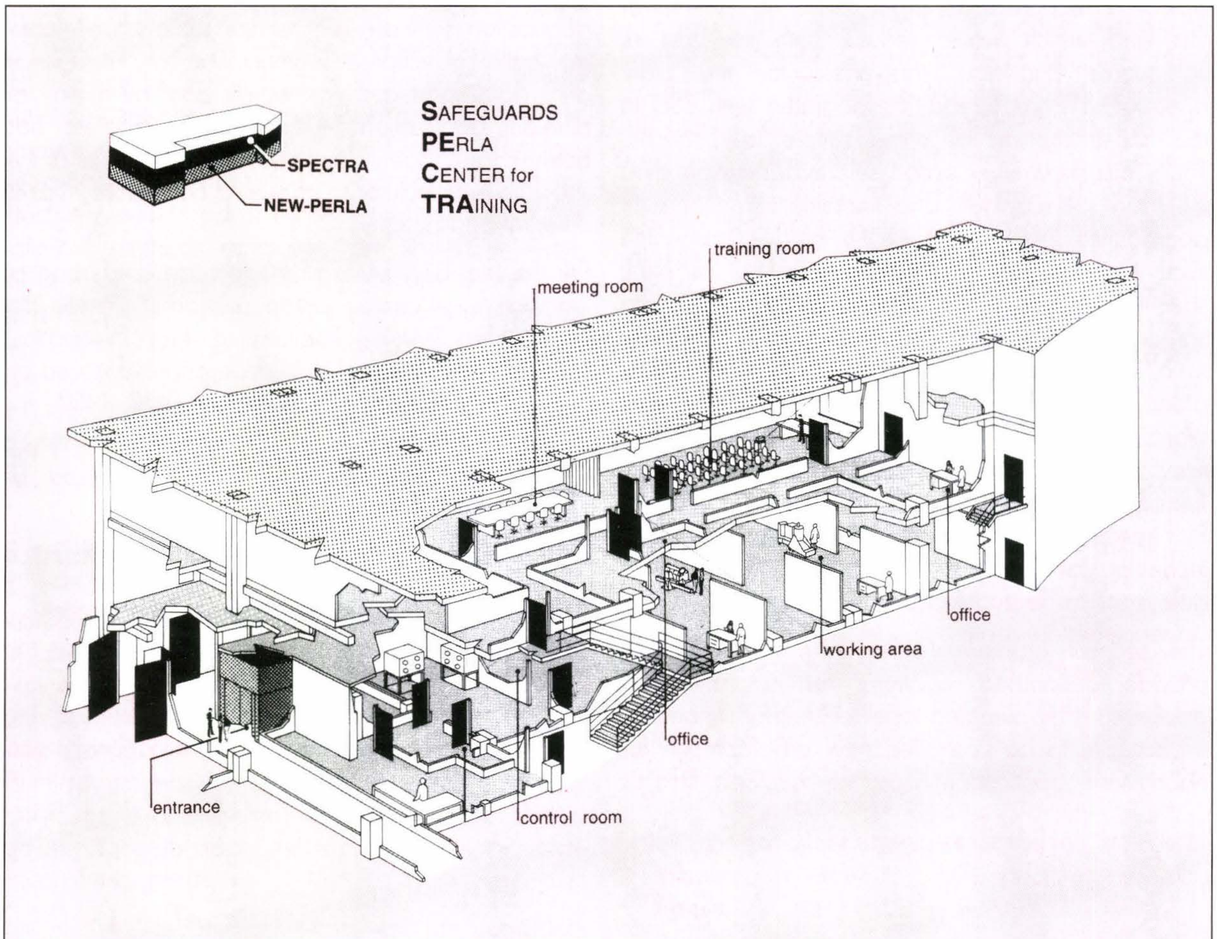

Fig. 3.8 Schematic view of NEW PERLA and PERLA training centre

1.3.4 PROGRESS IN TAME

The TAnk MEasurement Laboratory denoted with the acronym 'TAME" is currently being installed at the JRC Ispra (*Fig. 3.9*). The laboratory is in fact the ex-CALDEX (CALibration Demonstration EXercise) test facility which was installed in the TEKO facility of the KfK in Karlsruhe. Following agreements between KfK, GNS, IAEA and the JRC Ispra it was decided to transfer the facility to the Ispra site in 1991.

The re-installation of the whole equipment made it necessary to find an area with a useful height of approximately 15 metres. Towards the end of 1991 the structural design was completed and accepted and a contract was issued for installation. The basic structure was completed by the end of February 1992. By the end of 1992 the hydraulic and pneumatic circuits were completed and tested. A computerised process control, data acquisition and storage system has been purchased, installed and commissioned.

Fig. 3.9 A view of the TAME laboratory

Activities of the TAME laboratory

Based on the considerations coming from the workshop on TAME held at Ispra in March 1992, the possible areas of TAME activities might include the following:

- Performance evaluation of instruments used at the input accountability tanks, such as:
  - pressure meters (manometers, electromanometers)
  - pneumatic system
  - direct level meters, based on various physical principles (e.g. electrical capacity, sonar, radar)
  - weighing systems (scales, load cells)
  - thermometers
  - air humidity meters
  - liquid volume meters
  - airflow meters
  - densitometers
- Comparison of the precision and accuracy achievable with different operating procedures, namely:
  - incremental vs. decremental tank calibration
  - stepwise vs. continuous tank calibration
  - effect on calibration of small vs. large incremental steps
  - calibration with plain water vs. high density liquid
  - calibration with warm vs. cold liquid
  - measurement with vented vs. closed pipes
  - effect of the flow rate in the dip tubes
  - effect of homogenization procedures on the quality of the samples
  - effect of sampling procedures on the quality of the samples
- Physical effects of operating and environmental parameters, such as:
  - liquid loss by evaporation
  - effect on the total balance of hidden or

unmeasurable inventories (bottom of the tank, piping)
  - effect of ambient conditions, if not 100% under control (temperatures, pressures, humidities, etc.)
- Study of structural effects, such as:
  - tank deformation due to temperature
  - dip tubes aging (plugging, corrosion)
  - influence of the number, quality and location of the instruments (instrumental layout) on the quality of the results
  - influence of the homogenization method adopted and of the homogenization system layout on the quality of the samples
  - influence of the sampling method adopted and of the sampling system layout on the quality of the samples
- Authentication of measurements, data, samples
- Use of tracers for tank calibration and volume/mass measurements
- Data collection, treatment and interpretation:
  - performance of various data collection strategies (effect of the frequency of data collection, type of data measured, redundancy of measurements, etc.)
  - development and evaluation of statistical models
  - development and evaluation of models for the physical interpretation of data
  - development of criteria for extrapolation of models to different conditions
  - evaluation of operator's data interpretation models
- Training:
  - general training on the principles of mass and volume determination
  - training on the use of instruments
  - training on procedures for volume measurement and tank calibration.

References

[1] STI 1991 Annual Report, EUR 14803 EN

[2] Proceedings of the International Workshop on Calorimetry - JRC, PERLA Laboratory, Ispra (I), March 1992, EUR 14307 EN

[3] GUARDINI S. et al. - Assessment of Plutonium Calorimetry Performances at PERLA - Proceedings of the International Workshop on Calorimetry - Joint Research Centre - PERLA Laboratory, Ispra (I), March 1992, EUR 14307 EN

[4] VOCINO V., BINDA F., CARAVATI G., D'ADAMO D., FARESE N., MAUCQ T., REMORINI B. - Combined use of neutron correlation technique, calorimetry and gamma spectroscopy for the non-destructive mass determination of the Plutonium isotopes in nuclear fuel - Proceedings of the International Workshop on Calorimetry - JRC Ispra, March 1992, EUR 14307 EN

[5] BONDAR L., D'ADAMO D., DIERCKX R., HAGE W., HUNT B.A., PEDERSEN B., SCHILLEBEECKX P., VOCINO V. - Measurement facilities for radioactive waste at the JRC Ispra Establishment, 14th Annual ESARDA Meeting - Salamanca Spain, May 1992

[6] MERCIER M.T., HUCKINS R.J., ZALOKAR G.S., SMITH B.G.R., FRANKLIN M., BUTEZ M., EDELINE J.C. - A local Data Acquisition Subsystem for Plutonium Safeguards - Proceedings of the IEEE Nuclear Science Symposium, Fall 1991, Santa Fe, New Mexico, USA

[7] FRANKLIN M. - Automatic Sample Detection and Background Monitoring (ASDEM) - Technical Note n° 1.91.98, Commission of the European Communities, JRC Ispra, August 1991

[8] FRANKLIN M. - Error Propagation Consideration for Pu Mass Uncertainty - Technical Note n° 1.91.97, Commission of the European Communities, JRC Ispra, August 1991

[10] FRANKLIN M. - Statistical Models of NDA Measurements - Technical Note n° I.91.160, Commission of the European Communities, JRC Ispra, December 1991

[11] ZUKOSKY P.A., HODGES J.K.,. HUCKINS R.J, MERCIER M.T., ZALOKAR G.S., KOSKELO M.J., SMITH B.G.R., SWINHOE M., FRANKLIN M., BUTEZ M., MARTIN-DEIDIER L., RICHARD J.P., TALLEC M., PETIT S. - A Host Computer System for Distributed, Unattended Plutonium Safeguards - Proc. of the 33rd Annual Meeting of the Institute of Nuclear Materials Management, Orlando, USA, 19-22 July, 1992

1.4 FUSION TECHNOLOGY AND SAFETY

In the frame of the Community's Fusion Programme the JRC as a technically fully integrated Associated Laboratory concentrates its efforts and contributions on the demonstration of the safety and environmental feasibility of fusion power.

The identification of the quality and quantity of source term under normal and abnormal operating conditions is indispensable as input for a reliable safety analysis by facility design teams as well as for providing confidence to Regulatory and Radiological Protection Authorities in their assessment of the acceptability of envisaged protection systems, for any future tritium burning device.

The activities refer to a common "Leitmotiv" which is:

- *Minimization of the hazard, i.e. of the radioactive inventory at any time of the life of the plant.*
- *Minimization of the routine releases, i.e. containment of the radioactive products during the normal operation and routine maintenance phases.*
- *Minimization of the accidental releases, i.e. containment of the radioactive products during possible accidents occurring during the operational and the shut-down phases.*
- *Minimization of the doses to the operators during the life cycle of the plant.*
- *Minimization of the wastes and of the doses to the population before and after the decommissioning of the plant.*

They intend to cope in a complementary role to other Associations with "non-functional specifications", i.e. with the problems associated with the behaviour of the plant outside the limits fixed for its normal operating conditions and with the side effects on the population of this new type of energy. As such it can be the most reasonable response to the social and political demand of safety assurance.

As the boundaries between functional and non-functional specification are not well defined, this "mission" of the JRC has to be understood in terms of a general orientation, and does not exclude involvement in functional problems, where economically convenient for existing investments, competences, etc.

The European Tritium Handling Experimental Laboratory (ETHEL) will become fully operational during 1993. Research in ETHEL will essentially focus on tritium by identifying its transfer mechanisms, mitigating its propagation, minimizing its dilution and optimizing its confinement by studying:

- *Loss mechanism such as adsorption/desorption rates, permeation rates, leakages and the effects of potential remedies like permeation barriers;*
- *Multiple containment systems and fluid clean-up concepts under normal and accidental conditions;*
- *Methods for solid waste handling, treatment, conditioning and eventual disposal;*
- *Tritium control, monitoring and surveillance over the whole concentration range during both normal/abnormal operating conditions and maintenance activities.*

These items are of specific and primary interest for the NET TRITIUM TECHNOLOGY PROGRAMME and in future for ITER. Following informal contacts with JET representatives, some research areas have been found of common interest and useful to establish and develop complementary activities in support of the JET objectives.

In view of tritium experiments in ETHEL and to provide information on the basic processes involved in order to define final process schemes, preparatory experimental studies are being performed at the JRC-Ispra laboratories using hydrogen and deuterium. Experiments performed in the frame of the various preparatory studies are described below.

1.4.1 HYDROGEN ISOTOPES - MATERIAL INTERACTION

Transport of hydrogen isotopes in fusion reactors

LIBRETTO 3 (Liquid Breeder Experiment with Tritium Transport Option) [1]

As part of the Liquid Metal Blanket Programme, the in-pile irradiation LIBRETTO series of experiments has been carried out within the European Fusion Technology Programme on Blanket Technology as a joint project* between the Institute of Advanced Materials (IAM, JRC-Petten), CEA Saclay and the Safety Technology Institute (STI, JRC-Ispra). The irradiations are performed in the High Flux Reactor (HFR) at Petten.

The aim of the irradiation of the LIBRETTO 3 capsules was the in-pile testing of the efficiency of proposed tritium permeation barriers. The experiment employed four bubbled capsules including one capsule without any permeation barrier coating (blank capsule). Of the three coated capsules, one was coated with Al_2O_3 (by pack cementation) performed by the CEA (France) while JRC-Ispra coated the other two with Al_2O_3 and TiC using CVD (Chemical Vapour Deposition). During the first cycle of the capsules irradiation, the purging lines of the STI capsules became plugged and remained in that state during the other cycles. These capsules were therefore considered as closed capsules. The last cycle was dedicated to produce a neutron radiography of the capsules to show where the plugs were and to allow some conclusions to be drawn about what occurred to them. The data obtained from the irradiation are being processed by the CEA to extract permeation barrier efficiencies. Preliminary results seem to show that pack cementation is a better technique than CVD for producing permeation barriers.

In parallel to the in-pile testing, identical capsules to those used for the irradiation are being tested in the cold laboratory of ETHEL with deuterium. This test serves as a reference point for comparative purposes since all experimental parameters are well controlled. The measurements of the permeation of deuterium through the capsules coated with Al_2O_3 and TiC (STI-capsules) have been carried out whilst the CEA capsule was in the process of fabrication. In the course of 1993 these capsules will be filled with Pb-17Li and the effectiveness of the surface layers in contact with the breeder material will be assessed by direct measurement.

Calculation of tritium inventory, permeation and recycling in fusion reactor first walls

A Ph.D. thesis report on "Hydrogen Isotopes Transport in First Wall Fusion Reactor Materials" (Salford University, UK) was completed under the supervision and in collaboration with JRC scientists.

This thesis is concerned with the development of a computer programme for the study of the behaviour of hydrogen isotopes in first wall materials used in fusion devices. The study is of particular relevance to the development of safe and economic nuclear fusion based on the deuterium tritium (D-T) reaction, where the hazard related to the radioactivity of tritium and its high cost of production must be taken into account.

The source of hydrogen isotopes in the first wall is due to the (unavoidable) interaction between the plasma and the materials facing the plasma. Subsequently, the hydrogen diffuses in the lattice and may be trapped in lattice defects caused by neutron irradiation. If a hydrogen atom reaches the first wall surface it combines with another atom to form a hydrogen molecule (surface molecular recombination). All these phenomena form the basis of the model for the transport of the hydrogen isotopes in the first wall of a fusion device. The mathematical formalism associated with the model is quite complex and a computer code (TIRP2D) has been developed for the simulation of the above processes.

The quantities calculated by this code for hydrogen isotopes in the first wall are: the inventory in the wall, the permeation through the wall and the recycling

* Tasks LMB-BAR-J and LMB-PER-J

from the wall surface facing the plasma. The variation of these quantities as a function of time is presented. The materials for the first wall, considered here, are metals including low-activation alloys.The calculations have been performed for two Tokamak configurations:

- a large device such as ITER, which is foreseen to work with long pulse periods;
- a compact field device as IGNITOR, which is characterized by a very small shot period followed by a long dwell period.

The results indicate that the quantities calculated vary by many orders of magnitude according to the first wall material and that a high inventory and/or high permeation of tritium can be expected for large devices with some materials.

From the standpoint of tritium inventory, permeation and recycling, vanadium is the worst choice of the here considered materials due to its very high tritium solubility and diffusivity. Tritium inventory and permeation in a V first wall are about 4 orders of magnitude higher than in a first wall made of AISI 316L. Consequently, the characteristic time of recycling is much higher in V than in 316L.

These properties of a V first wall should be similar in a first wall made of V-15Cr-5Ti which is a candidate first wall material due to its favourable activation properties.

Recycling of hydrogen isotopes from first wall materials of fusion reactors [2]

Recycling of hydrogen (tritium and deuterium) from plasma-facing components in a magnetic fusion device is widely accepted to be a crucial issue affecting the fuelling and the tritium inventory of the reactor. Consequently, an experiment has been designed to study the hydrogen isotopes implantation, uptake and release of AISI 316L, MANET, graphite, CFC, doped CFC and low Z materials under similar conditions to those of a fusion reactor (ITER), namely:

- the implantation at low energies (50 - 200 eV), high flux densities (10^{16} - 10^{17} atoms $cm^{-2}s^{-1}$) and at different temperatures;
- the measurement of hydrogen isotope (tritium) release rates as a function of time and characteristic recycling times at different ion energies, particle flux densities and target temperatures.

The main part of the facility is a plasma simulator that can produce ion flux densities and energies as specified above. This cold installation is used as a test bed for a tritium compatible one being installed in a glove box of ETHEL (see subparagraph on ETHEL-001). Identification and optimization of the process control parameters, i.e. flow, gas composition and temperature, prior to the tritium operation is being carried out, as well as the fine tuning of the control procedure. The dependence of the positive ion fraction of the plasma (D^+, D_2^+, D_3^+) on the gas pressure has been identified and a parametric study is on the way. The fraction of positive ion species gives both: the total particle flux impinging on the target and its energy distribution. These two parameters are needed in order to properly define the recycling flux.

Solubility, diffusivity and permeability in fusion reactor materials

MANET (MArtensitic for NET) [3,4]

The martensitic steel DIN 1.4914 (MANET) is one of the two candidate materials for the first wall and structure of the next step fusion reactor ITER and for the demonstration power reactor DEMO.

The martensitic samples consisted of one cylinder of 20 mm diameter and 100 mm length and 4 cylinders of 7 mm diameter and 60 mm length. They were produced from semifinished rods supplied by KfK Karlsruhe of the following composition (wt%): C 0.13, Cr 10.6, Ni 0.87, Mo 0.77, V 0.22, Nb 0.16, Si 0.37, Mn 0.82, S 0.004, P 0.005, B 0.0085, N 0.020, Al 0.054, Co 0.01, Cu 0.015, Zr 0.053, Fe balance. The samples underwent the following heat treatment in order to produce a fully martensitic phase: heating at 1243K for 2 hours, austenizing at 1348K for 0.5 hours, quenching to room temperature, tempering at 1023K for 2 hours, slow cooling to room temperature.

The experimental method used was a gas release technique. A single experimental run consisted of two phases: first the sample was loaded with H_2 or D_2 at a given temperature and pressure until saturation was reached and then, after a short pump down to less than 10^{-4} Pa, the pressure increase due to gas release from the sample and from inner walls was measured in a calibrated volume until a new equilibrium was obtained. Each run was followed by a blank run under the same experimental conditions but without a sample, and the contribution of the inner wall surface thus obtained was then subtracted from the total pressure increase to give the net contribution due to the sample alone. The solubility was determined from the final pressure in the calibrated volume and the diffusivity from the pressure vs. time curve by a non-linear least squares fitting process to a diffusion controlled gas release model.

Sieverts' constants and diffusivities of hydrogen isotopes in MANET are presented in ***Figs. 4.1*** and ***4.2***. The tritium values were calculated from the measured ratios of hydrogen and deuterium solubilities and diffusivities by application of quantum-statistical partition functions to the solution and diffusion process. Further heat treatment of the martensitic samples at 810K or 900K showed nearly no change of hydrogen solibility, but a strong decrease of hydrogen diffusivity (curve 4 and 5 in ***Fig. 4.2***). On the contrary, hydrogen diffusivity was nearly not changed, when the additional heat treatment was carried at temperatures below 740K.

The experimental data can be presented by the following Arrhenius equations, where T is given in K.

Solubility of H2 in MANET:
K_s / (at. fr. $Pa^{-1/2}$) = 9.83.10^{-6} exp (-3740/T)

Diffusivity of H_2 in "virgin" MANET:
D / (m^2 s^{-1}) = 9.85 .10^{-8} exp (-1540/T)

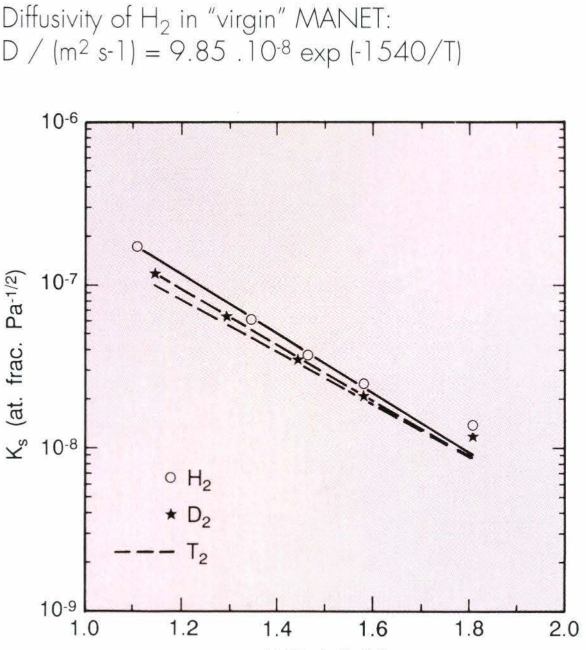

Fig. 4.1 Sieverts' constant of hydrogen isotopes in MANET

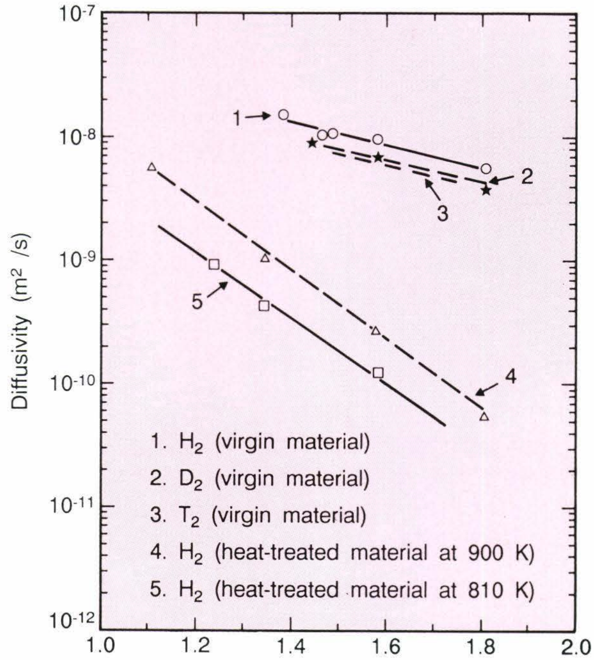

Fig. 4.2 Diffusivities of hydrogen isotopes in MANET

Diffusivity of H_2 in MANET, heat treated at 900 K:
D / (m^2 s^{-1}) = 1.56 . 10^{-5} exp (-7030/T)

Diffusivity of H_2 in MANET, heat treated at 810 K:
D / (m^2 s^{-1}) = 6.96.10^{-6} exp (-7270/T)

The processes which can reduce the measured diffusivity of hydrogen and therefore provide an explanation of the above results in a solid material are:

a) Formation of an oxide layer
b) Change of structure
c) Trapping
d) Decrease of density of internal surfaces
e) Decrease of effectiveness or blocking of internal surfaces

It could be shown that the processes b) and c) could be ruled out as the cause of the change in the observed diffusivity and that the processes a) and d) can explain only a diffusivity decrease by a factor of 2.

The observed strong decrease of hydrogen diffusivity and increase of activation energy of diffusion after heat treatment could indicate a change of predominant diffusion mechanism from diffusion along grain or lath boundaries to volume diffusion. It should be noted that in a martensitic steel there is a much greater density of boundaries than in ferritic or austenitic steels. Diffusion and segregation of interstitial C or N and of impurities to grain or lath boundaries during heat treatment can block them for preferred hydrogen diffusion. Evidence for this hypothesis can be provided by Auger-analysis of broken specimens of "virgin" and heat treated MANET, which will be performed in February 1993 at the Perkin Elmer Laboratory at Munich.

Influence of coating on permeation [5]

The main aim of the work is to produce tritium permeation barriers by the application of surface layers on materials of interest to fusion research. The experiments take the form of the measurement of the permeation rate of deuterium gas through metallic specimens both with and without permeation barriers and the characterization of the surface layers by a variety of techniques. At the beginning of 1992, the main experimental activities were transferred to the cold laboratory of ETHEL (European Tritium Handling Experimental Laboratory).

In 1992, the permeation of deuterium through a wide range of materials was investigated. The samples used for the investigations were in both disk and tube forms. Measurements were taken in the temperature range of 470 to 750K (which is the temperature region of most interest for fusion reactor structures such as the breeder blanket) and using deuterium pressures in the range 1-100 kPa. Uncoated samples investigated include MANET (a modified DIN 1.4914 martensitic steel which is a candidate structural material for ITER), 316L stainless steel and the molybdenum alloy TZM (a possible fusion reactor first wall material). The measurements on TZM yielded values for the diffusivity, solubility and permeability of hydrogen isotopes in this alloy, which could constitute useful design data for ITER and other fusion machines. For example, it was shown that the alloying elements in TZM, Ti and Zr, cause a significant amount of trapping of hydrogen isotopes.

In addition, the following coating systems have been investigated: TiC layers of various thicknesses, a bi-layer of TiN on TiC, and Al_2O_3 on TiN/TiC. All these layers were produced by the Chemical Vapour Deposition (CVD) technique.

Some results for the permeation measurements, plotted in the Arrhenius manner, are shown in ***Figs. 4.3*** and ***4.4***. ***Fig. 4.3*** shows the effects of 1 or 2µm thick CVD TiC layers on 0.1mm thick TZM, whilst ***Fig 4.4*** shows the effects of TiN+TiC and Al_2O_3+TiN+TiC layers on 1mm thick 316L stainless steel. In all cases it was found that the CVD technique could produce only moderately effective permeation barriers, at best reducing the permeation rate by around an order of magnitude compared to uncoated samples. At a temperature of 673K, the intrinsic permeability of TiC was estimated to be 6000 times less than that of 316L stainless steel. It also appeared (as shown in ***Fig. 4.4***) that the CVD Al_2O_3 played little or no part in reducing permeation, presumably due to porosity.

Further data on the usefulness of TiC as a permeation barrier was made available with the presentation of a Ph.D. thesis at the Open University, Oxford, entitled "Hydrogen Interactions with TiC-coated

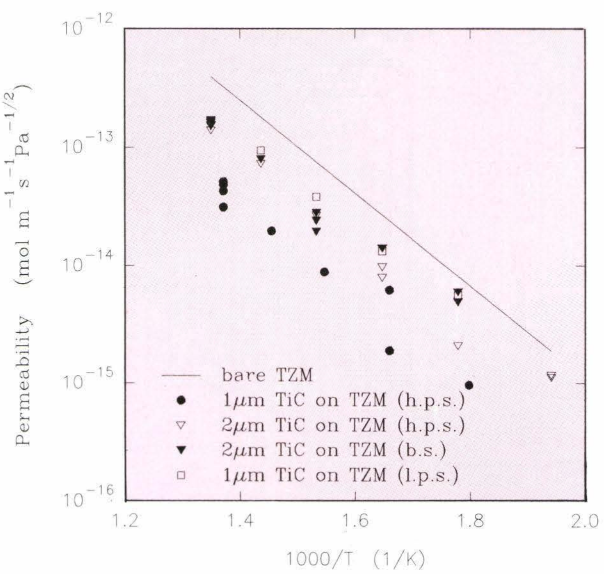

Fig. 4.3 An Arrhenius plot showing the effect of CVD layers of TiC on the permeation of deuterium through TZM

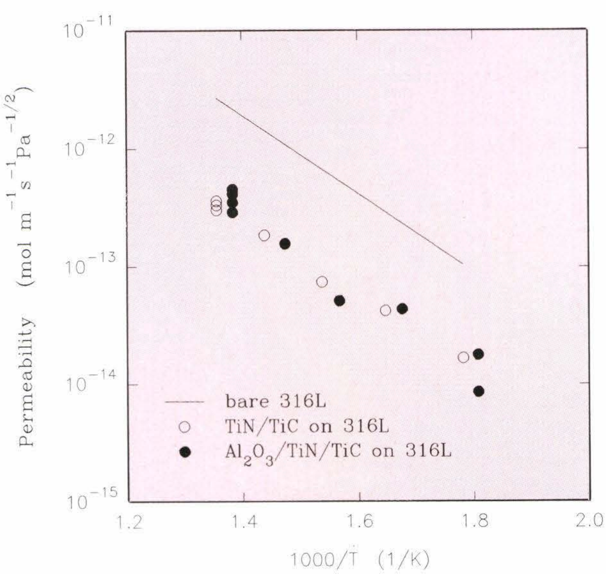

Fig. 4.4 An Arrhenius plot showing the effect of CVD layers of TiN + TiC and Al2O3 + TiN + TiC on the permeation of deuterium through 316L stainless steel

Metals for Fusion Technology Applications". The work for the thesis was carried out under the supervision of staff from the Safety Technology Institute and involved a considerable input from JRC Ispra personnel. The prime aim of the thesis was to determine whether TiC coatings deposited on metals by CVD could affect the rate of absorption and desorption of hydrogen isotopes. Another goal of the work was the development and testing of simple models for the experiments using the following parameters: diffusivity D, Sieverts' constant K_s, and surface reaction rates k_1, k_2. Both aims were closely linked to the use of tritium in fusion reactors. The thesis describes the two basic techniques employed. At Ispra measurements were performed by monitoring the hydrogen pressure increase in a closed vessel after loading the sample with hydrogen to equilibrium with a pressure in the range 1-100 kPa and then evacuating the chamber. Measurements were taken at temperatures in the range 673-873 K. At Oxford, a pressure modulation technique was employed to obtain similar data. The substrates used were 316L stainless steel and TZM. Both methods revealed that the coatings resulted in a marked increase in the time to reach equilibrium but the hydrogen solubility was not significantly affected. Very low diffusivity values were obtained for hydrogen in TiC. The measurements at Ispra gave the following equations for the diffusivity and surface rate constants:

$D=1.08 \quad \times 10^{-12} \quad \exp(-6800/T) \quad (m^2\, s^{-1})$
$k_1=4.1 \quad \times 10^{-8} \quad \exp(-4560/T) \quad (mol\, m^{-2}\, s^{-1}\, Pa^{-1})$
$k_2=6.9 \quad \times 10^{-12} \quad \exp(-1595/T) \quad (m^4\, mol^{-1}\, s^{-1})$

whilst at Oxford the following relations were obtained:

$D=9.6 \quad \times 10^{-12} \quad \exp(-6010/T) \quad (m^2\, s^{-1})$
$k_1=5.4 \quad \times 10^{-5} \quad \exp(-10250/T) \quad (mol\, m^{-2}\, s^{-1}\, Pa^{-1})$
$k_2=2.9 \quad \times 10^{-9} \quad \exp(-5070/T) \quad (m^4\, mol^{-1}\, s^{-1})$
$K_s=1.35 \quad \times 10^{2} \quad \exp(-2590/T) \quad (mol\, m^{-3}\, Pa^{-1/2})$

The results suggested that TiC could impede hydrogen permeation through steel more effectively in surface-limited regimes. TZM was also studied and was considered as a possible permeation barrier itself. Trapping was quantitatively detected in TZM.

Towards the end of 1992, it was discovered that a much more effective permeation barrier could be formed by an aluminizing process. MANET discs were coated with aluminium by a plasma-spraying process and subsequently heat-treated to cause interdiffusion of surface and substrate. The result was a surface coating consisting of an iron-aluminium intermetallic that reduced the permeation rate through the steel by up to 3 orders of magnitude. ***Fig. 4.5*** shows an Arrhenius plot of the permeation rate through the aluminized layer compared to uncoated

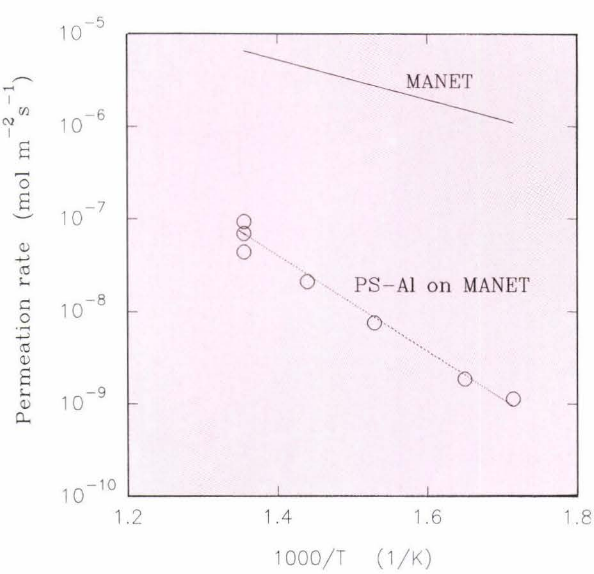

Fig. 4.5 An Arrhenius plot showing the effect of an aluminized surface layer on the permeation rate

material. The interdiffusion process ensured that the layer is highly adherent to the substrate. In addition, the particular route taken to produce the aluminized layer, which involved combining the heat-treatment to cause interdiffusion with the normal tempering procedure recommended for this type of steel, meant that the steel was kept in its optimum from the point of view of microstructure. Thus, an extremely effective permeation barrier has been formed on steel without adversely affecting the mechanical properties of the underlying material. This finding represents an important advance in the search for a practical solution to the problem of tritium permeation. This work was carried out in collaboration with the Institute of Advanced Materials at Ispra.

During 1992 the permeation apparatus was further modified to enable permeation measurements to be made on the stainless steel capsules designed for the LIBRETTO 3 experiment (see subparagraph on LIBRETTO 3).

Tritium permeation through engineering components [6]

Permeation of tritium through components of a fusion reactor poses a safety problem. Although studies exist of the permeability of hydrogen through several materials of interest for fusion technology, little data are available on the permeation of tritium through engineering components in a real tritiated environment, where mechanical stresses might be of importance when quantifying the tritium permeation through them. The object of this experiment is to obtain data on the permeation of tritium through engineering components typical of those to be used in a large-scale fusion reactor. Presently the study is focused on Inconel 625 bellows which are engineered components employed at JET to accommodate thermal movements, allow fine alignment of components, reduce mechanical shocks and permit large movements within some systems.

An experimental facility has been constructed (see Institute 1991 Annual Report) and operated to obtain the required data on the tritium permeation rate. Parallel to this, a theoretical model to interpret the data has been developed. The simple model developed is a surface-limited model, i.e. the diffusion, much more rapid than the recombination at the surface, cancels the concentration gradient within the membrane. The permeation rate is proportional to the pressure and independent of the thickness, depending purely on the surface parameters. The permeability of the bellows has been obtained by fitting the experimental data to the model ***Fig. 4.6***. Although good fits are obtained, the simplified surface model assumption that the upstream pressure is kept constant throughout the length of the experiment is not satisfied by the experimental setup since no continuous feed of tritium is applied to the upstream volume (bellows volume). However, the use of this approximation is justified in a time region where the upstream pressure (P_h) is much larger than the downstream pressure (P_l), i.e. the tritium loss from the high pressure side by permeation is insignificant. It should be kept in mind that the upstream pressure is of the order of 10^{-2} Pa (250 mCi m^{-3}) while the downstream pressure at the highest temperature investigated is of the order of 10^{-6} Pa (100 μCi m^{-3}), that is P_h is four orders of magnitude larger than P_l.

Two sources of error in the determination of the permeability have to be considered. One is the error in the measurement of the amount of tritium permeating through the bellows, i.e. error due to the ionization chamber, which in the range of some hundreds of μCi m^{-3} is around 30%. The other is the error in the temperature measurements, due to the fact that the temperature of the flexing bellows cannot be directly measured, but only at the two metal end flanges. This error depends on the steepness of the temperature dependence of the permeability although a mean error of 20% has been assumed. This results in a total error of 40% which may be slightly pessimistic.

The experimental results, ***Fig. 4.7***, show that flexing of the bellows produces a permeation rate that is about an order of magnitude higher than in the case of stationary bellows. This is due to cracks which develop in the oxide layer covering the surface during the flexing. However, from the safety point of view, the permeation rate is around 100 times lower than the one expected in the case of the diffusion limited regime and agrees well with the values obtained by Zarchy and Axtmann [8] who studied the permeation through 304 stainless steel at ultra-low pressures (in the range of 10^{-5} to 10^{-7} Pa).

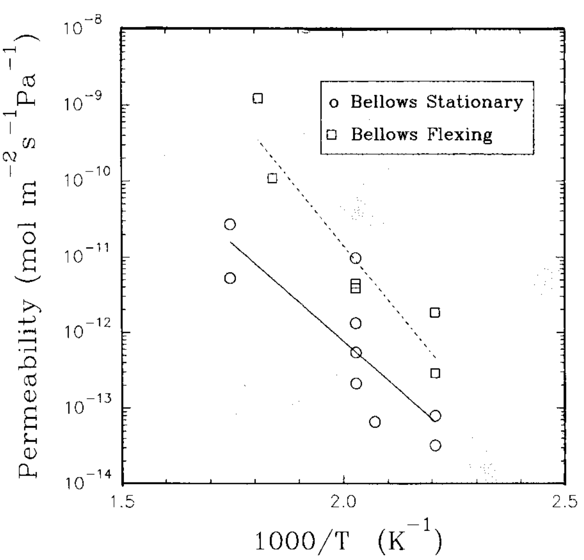

Fig. 4.6 Temperature dependence of the permeability of tritium through the inconel bellows

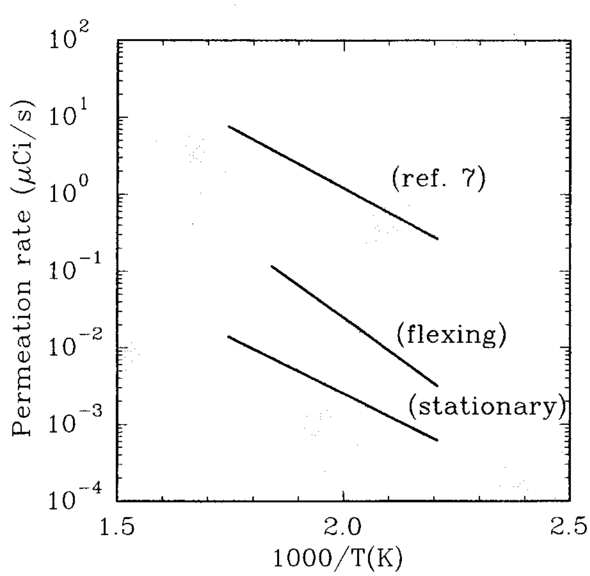

Fig. 4.7 Permeation rate of tritium through the inconel bellows compared with the values obtained by diffusion-limited permeability [7]

1.4.2 GAS SEPARATION PROCESSES ON SOLID SUBSTRATES

Development of new sorbent materials

The aim of this research is to prepare, characterize and select materials for use as substrates in gas-solid separation processes. The preparation of substrates includes their modification using ion exchange techniques and pore size engineering. In particular, new types of substrates made by "pillaring" clay materials with alumina polymers were studied.

In recent years there have been stimulating reports on the possibility of inserting thermally stable inorganic pillars between the layer of the structure of smectite clays, in order to modify their texture, surface area, porosity, and acidic properties. The intercalating species are polyoxidations of Al or other elements (Zr, Mg, Fe, Cr, Zn, Ti...). Depending on the intercalated complex cations (the pillar) the interlayer spacing can be varied almost continuously, giving rise to a large number of substrates with a tailored porosity larger than in zeolites. These products are generally called PILC.

The fluid separation group has experimented with PILCs obtained by pillaring the naturally occurring phyllosilicates montmorillonite (English China Clays - Mineral Colloidal BP) with the Al_{13} polyoxidation: $Al_{13}\ O_4\ (OH)_{24}\ (H_2O)_{12})^{7+}$. Its structure is known to consist of 12 octahedral Al ions sourrounding an aluminum ion according a tetrahedral coordination. The basic composition of motmorillonite can be represented by the formula: $Na_x\ (Al_{2-x}\ Mg_x)\ (Si_4O_{10})\ (OH)_2\ zH_2O$, where x depends on the degree of substitution of Al by Mg (Fe^{2+} can also substitute Al) and z on the degree of hydration. Aluminum, or its substitute, has an octahedral coordination, while silicon has a tetrahedral coordination. A stereotype of the motmorillonite structure before and after the introduction of the pillars is represented in ***Fig. 4.8***.

The optimized procedure for preparing the PILCs was found to be the following: degree of hydrolysis of the pillaring solution OH/Al = 1.85; final pH = 4.64 (at 277K) and aluminum concentration [Al] = 0.07M. After refluxing the pillaring solution, clay was added (2.3% clay, 3 mol Al/g clay) at 340K. The filtered product was washed, filtered and dried at 390K. The calcined PILC (770K) was treated with ammonia vapour to increase the cation exchange

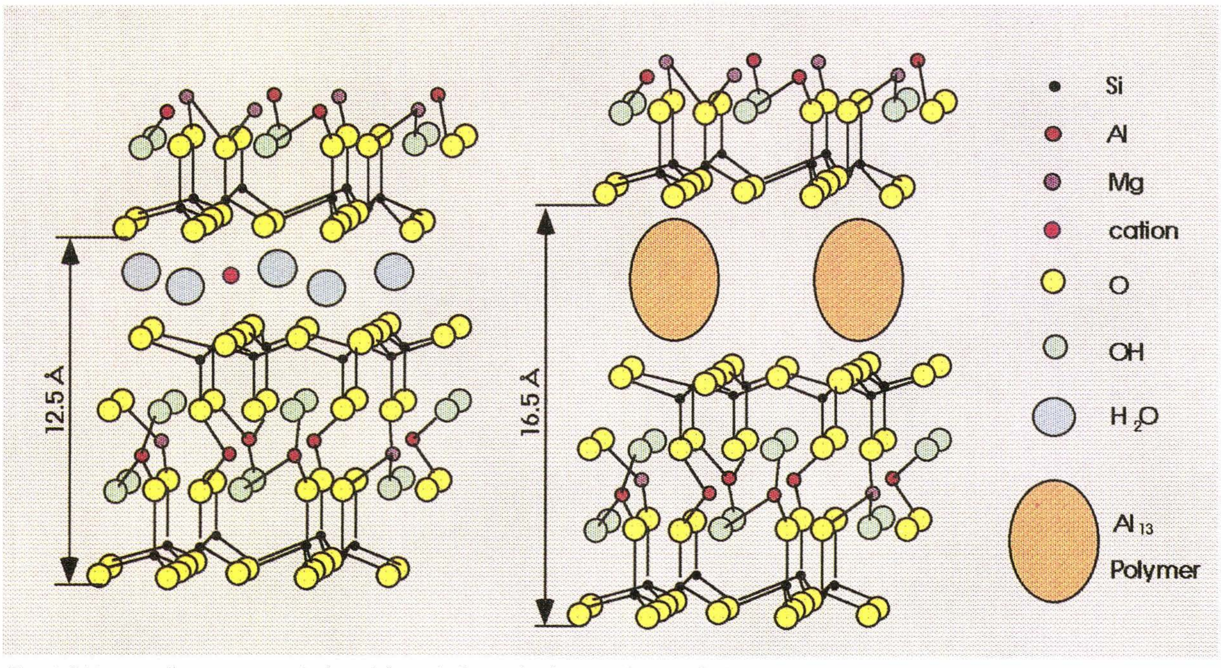

Fig. 4.8 Montmorillonite structure before (left) and after (right) the introduction of Al_{13} polymers

capacity (CEC) and then exchanged with a 0.2M NaCl solution. Activation at 520K under ultra high vacuum for 24 hrs yielded a porous substrate (Na/Al-PILC5) with a Dubinin surface area of 246 m^2/g and a micropore volume of 0.14 cm^3/g.

The optimization was performed with respect to the following synthesis parameters: OH/Al ratio, Al concentration, Al/g clay, pH, exchange and calcination temperature. The micropore volume was used as the principal response parameter. *Fig. 4.9* reports a summary of the results of the parameter study. The effect of calcining in air has been studied using X-ray diffraction (*Fig. 4.10*). The interlayer distance (d001) tends to decrease, while the diffraction peak width increases with temperature. Below 670K rehydration restores the original state of pillar. On the other hand, in a region of 620-670K the peak width increases, which indicates that some domains have been irreversibly transformed. Above 670K a more general reaction seems to occur which is completed near 720K. Differential thermal gravimetry (DTG) of the sample confirms this transformation.

Adsorption isotherms measured on Na/Al-PILC5 are reported in *Fig. 4.11*. The pore size distribution calculated by means of a specially developed and validated model [9-11], using different pore geometries, is reported in *Fig. 4.12*. The analysis of the X-ray diffraction pattern gives an interlayer spacing (d001) of 16.5 Å and confirms that the slit geometry is the most suitable for evaluating the pore distribution structure of this pillared clay.

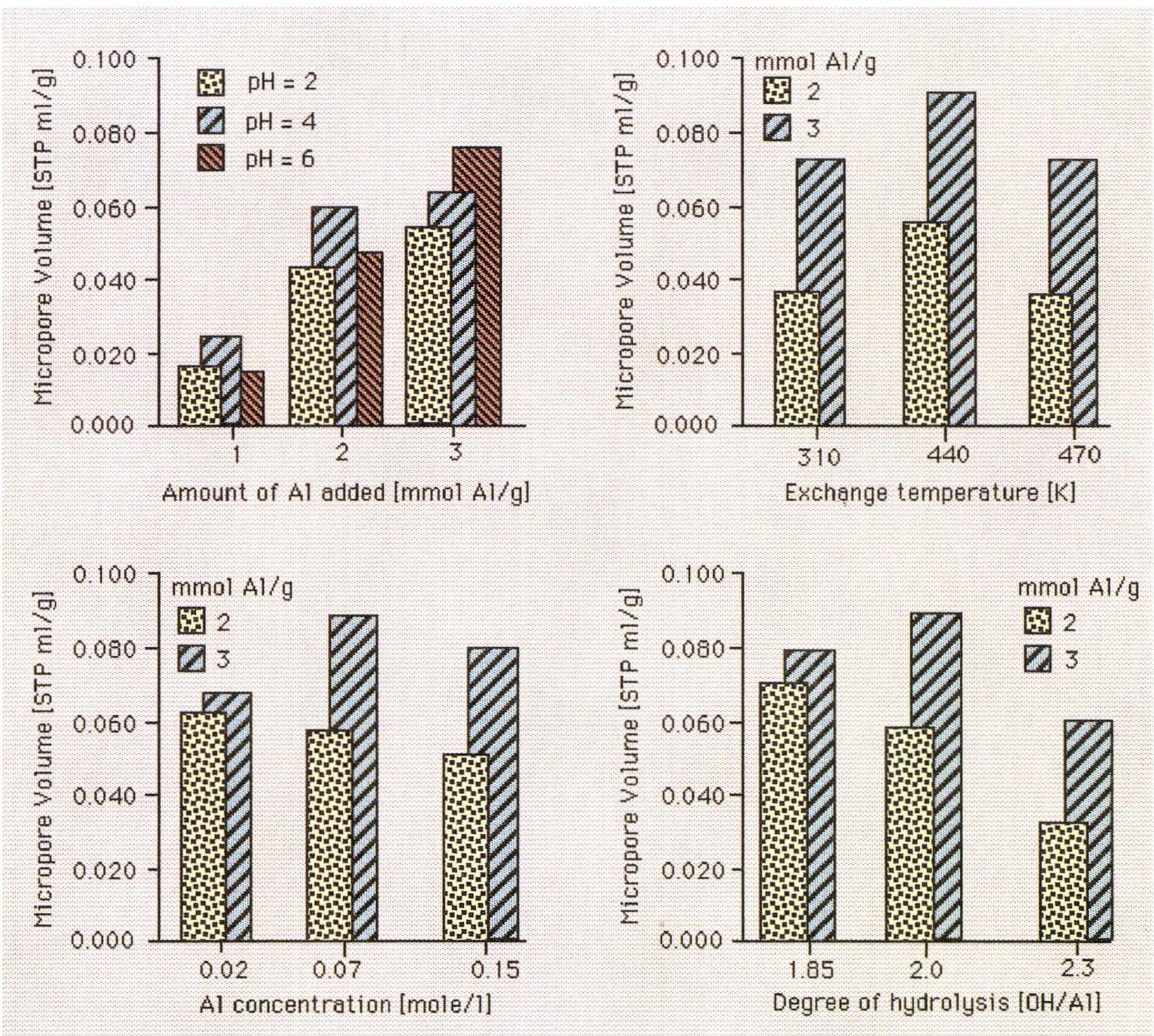

Fig. 4.9 Influence of different parameters on the PILC micropore volume

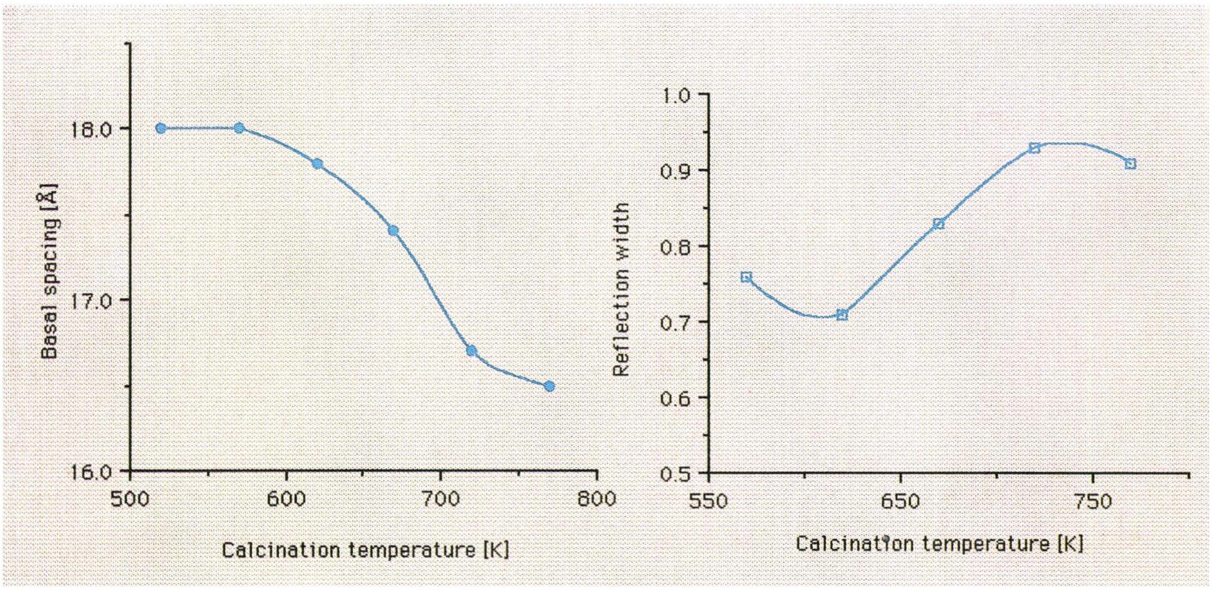

Fig. 4.10 Variation of the PILC basal spacing and reflection witdh versus calcination temperature

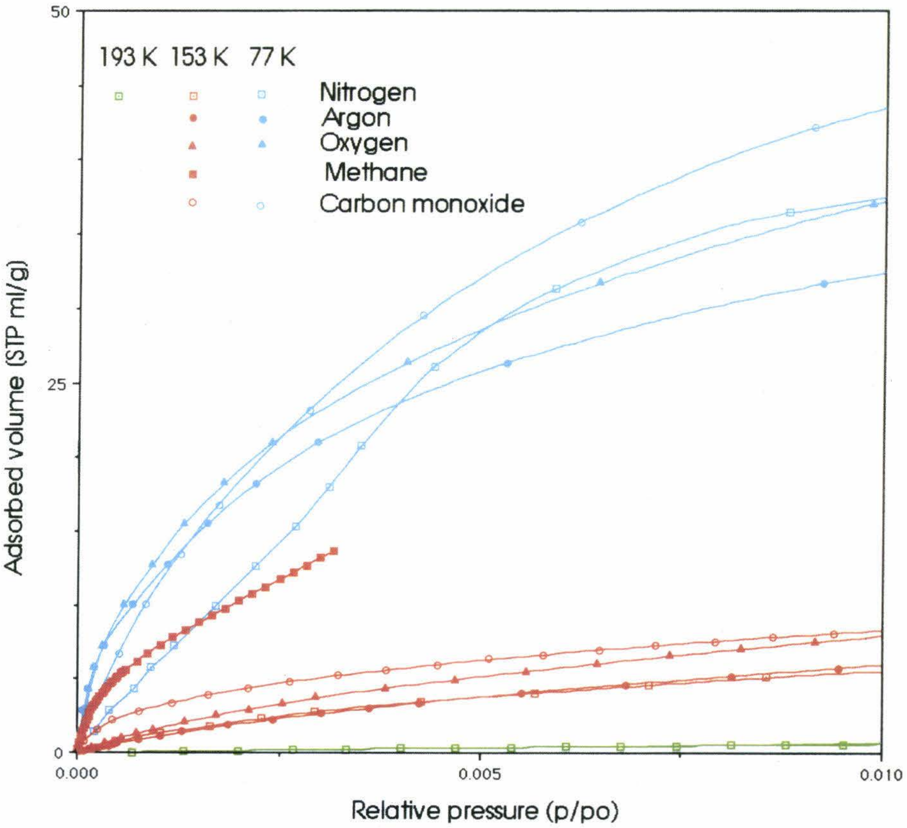

Fig. 4.11 Adsorption isotherms for pillared montmorillonite Na/Al-PILC5

This substrate is well ordered and provides a considerable micropore volume yielding a good adsorption capacity in the low temperature range. Considering that hydrogen is hardly adsorbed at 153K, this material is very promising for some applications only; such as the cryosorption of impurities from the plasma exhaust stream of fusion reactors. However, the lack of kinetic adsorption differences between the different gases leads to the conclusion that the pore size is too large for molecular sieving effects. This is to be expected, considering the large interlayer distance. Moreover at higher temperatures the adsorption capacity becomes low in comparison with zeolite materials. These effects can be attributed to a lack of localized adsorption in the material. Modification of the Al-PILC with other cations: Li^+, K^+, Cs^+, Mg^{2+}, Ca^{2+}, Sr^{2+} and Ba^{2+}, should give a substrate with increased localized adsorption. Work in this direction is in progress. The work is performed in collaboration with Prof. E.F. Vansant from the University of Antwerp.

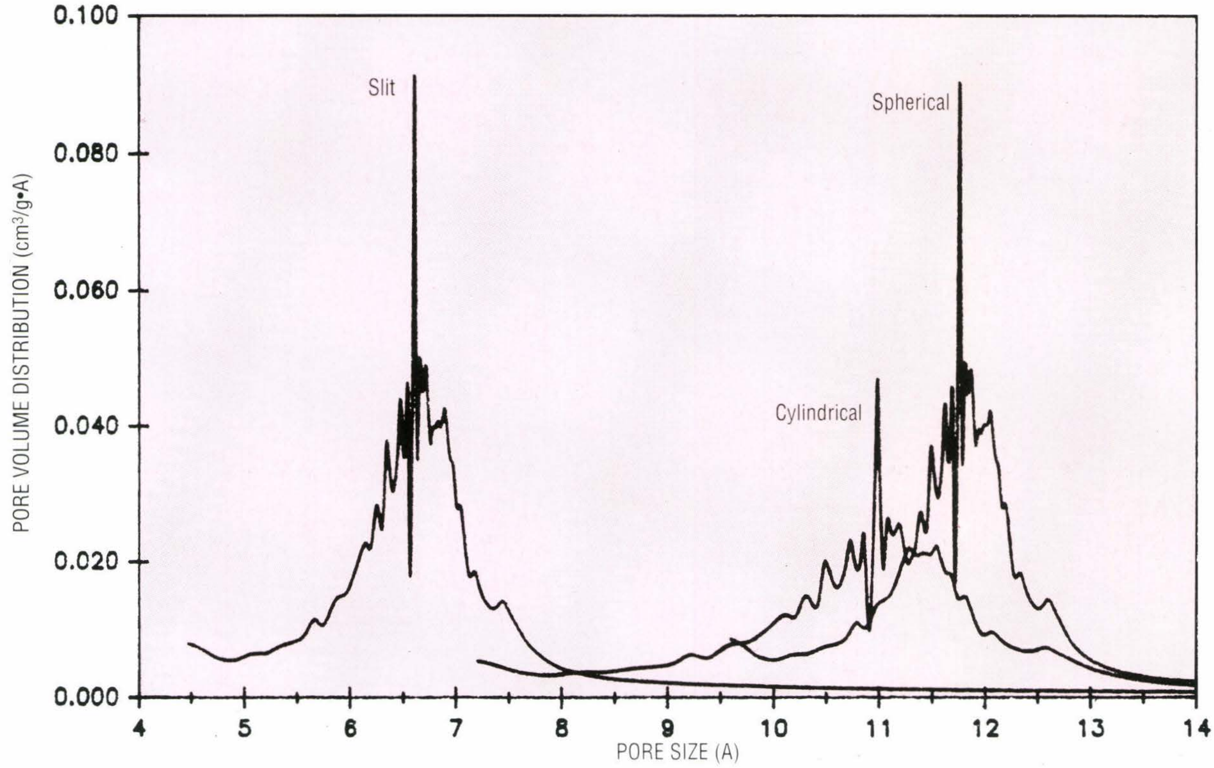

Fig. 4.12 Micropore size distribution of Na/Al-PILC5 from methane isothermal data at 77K for each model

Isotopic separation of hydrogen by gas chromatographic techniques

This research activity has three objectives:

a) Diagnostic analysis of the six isotopomeric forms of hydrogen (H_2, HD, HT, D_2, DT, T_2).
b) Separation and analysis of a DT/T_2 mixture in less than 10 minutes.
c) Separation of large quantities of $(D/T)_2$ from hydrogen by displacement gas chromatography.

to a) A large experimental programme has been concentrated on different types of zeolites of the mordenite family. For separating a ternary mixture of $H_2/HD/D_2$ the best results have been obtained using as sorbent a Ca (45%) Na-Mordenite small port, specially modified by the Grande Paroisse Co., Montoir, France. Separation and resolution factors measured using such a substrate are reported in ***Table 4.1***. A large number of tests using other materials [12], have shown that separation and resolution factors between the isotopes inverse sharply with the Henry constants of H_2, HD, D_2 and with their desorption rates. A highly crystalline Na-Mordenite large port also showed high desorption rates combined with high adsorption capacity. In

Table 4.1 Experimental results of chromatographic separation of a H_2/D_2 binary mixture on Ca (45%) Na-mordenite (2 mm long, 2.1 mm i.d. column) performed at a constant He carrier flowrate (40 ml/min)

T(K)	tH_2(min)	tD_2(min)	SH_2/D_2	RH_2/D_2
175	4.91	6.93	1.41	3.30
178.5	4.26	5.91	1.39	3.12
182	3.72	5.05	1.36	3.11
183	3.53	4.69	1.33	3.26
193	2.45	3.20	1.31	2.94
203	1.84	2.10	1.14	2.43

Fig. 4.13 the separation factors of the two sorbents can be compared.

Literature results indicate that a separation factor of 4 is required to obtain satisfactory separation of the six isotope combinations. However, such a factor has only been reported for low temperature adsorption, which takes 4-5 times longer than in our tests. The best separation factor we have so far obtained is 3.5, but we expect that further modifications will increase this value to more than 4, without increasing the retention times. This should enable to achieve full isotopic analysis of the gas stream within 12 minutes.

to b) Separation of H_2 and D_2 has been obtained within 3 minutes using a Na-Mordenite large port modified by Ca^{2+} and Ni^{3+}. This means that the DT/T_2 separation can be easily accomplished in less than 10 minutes at around 200K by satisfying the requirements of the JET.

to c) The objective of this research is the separation of large quantities of $(D/T)_2$ from H_2 by using displacement gas chromatography. The feasibility of this procedure is tied essentially to finding sorbents with high capacity of adsorption and desorption of the hydrogen isotopes. Preliminary research has shown that some types of mordenite or modified mordenite as well as Y type zeolites have the required properties.

Cryosorption on molecular sieves or alternative cryosorbent*

The cryosorption of impurities from the plasma exhaust stream is an alternative to permeation through a Pd/Ag membrane. However, it appears worthwhile to consider the integral cryosorption process not only as a potential alternative, but also a backup process to permeation which could be integrated into the ITER design.

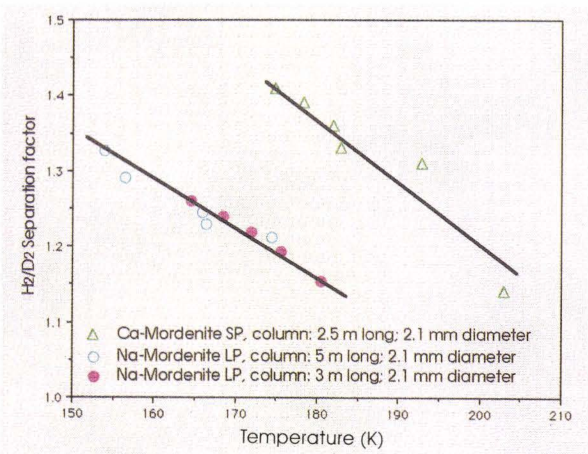

Fig. 4.13 Separation factor for a H_2/D_2 binary mixture as a function of temperature on different adsorbents

Consequently new substrates must be developed which should be able to satisfy the following requirements:

- selectivity for the various impurities potentially present in the plasma exhaust gas stream and capability of separating such impurities to non-detectable levels;
- insensitivity to poisoning effects and resistance to material degradation due to the chemical and physical phenomena (embrittlement, radiation damage);
- complete reversibility for all species in the adsorption-desorption cycles and exclusion of irreversible tritium trapping in the substrate matrix;
- minimization of transient tritium inventories during the adsorption cycles of impurities;
- simplicity of operation, compactness of components and installation, ease of scaling-up and reliability and cost effectiveness of the proposed process.

In the light of these requirements, which correspond to the objectives of the NET Task (TEP 1-3), the use of materials such as 5Å zeolites has to be excluded. The research work at Ispra is aimed at developing and testing new types of substrates prepared by modifying the type of cations in the structure and basic framework.

Many of these objectives can already be met using mordenites [12-13], both small and large pore: a complete separation from some specified impurities of the plasma exhaust can be obtained at temperatures of 160K. On the basis of the experimental results so far obtained a flow-sheet [14] for the separation of impurities from the plasma exhaust stream has been developed. This flow sheet will be tested in the frame of an experiment (ETHEL 005) to be performed in ETHEL. The experiment is intended to purify 45 dm^3 of a DT contaminated gaseous stream within 8 hours. A schematic design of the Clean-up System (CUS) apparatus, and the related column operating schedule are shown in *Figs. 4.14 & 4.15* respectively. This work is performed in collaboration with Prof. A. Viola from the University of Cagliari.

* NET-TASK TEP 1-3

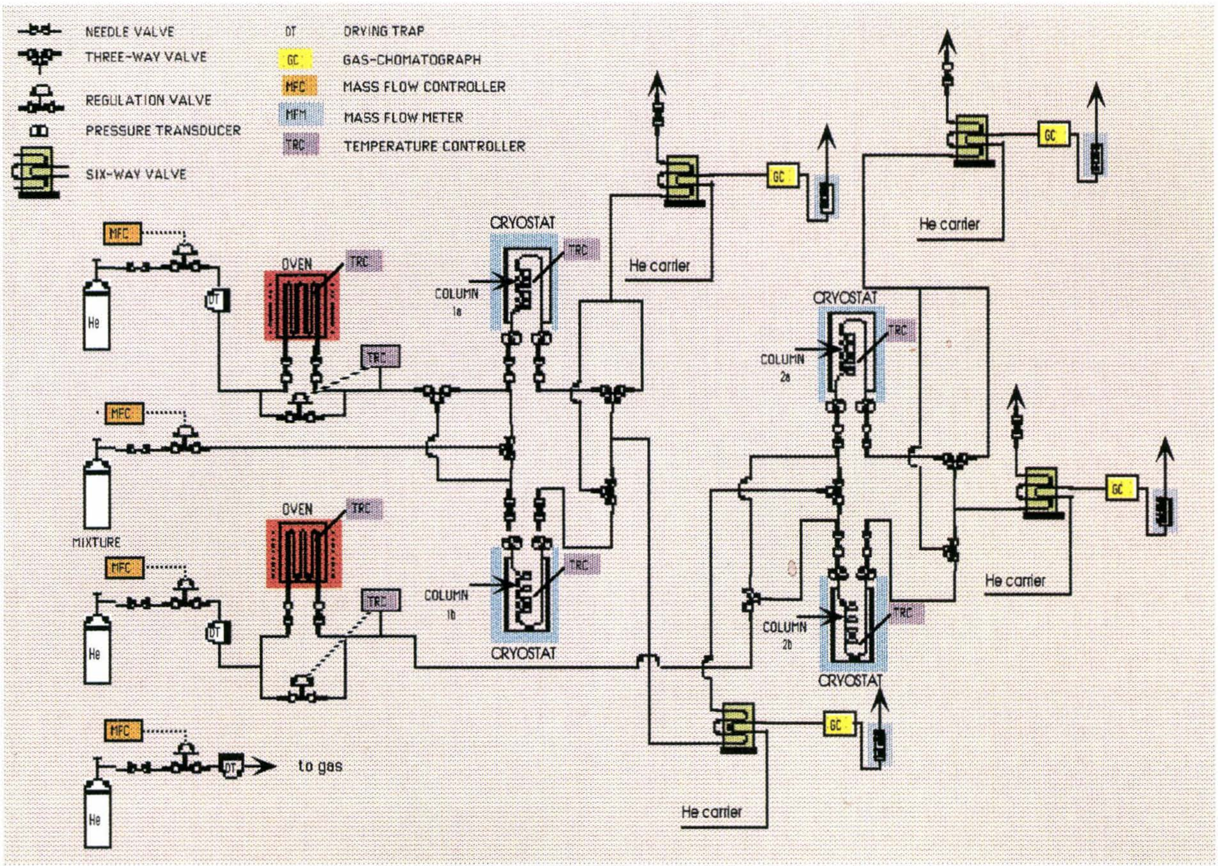

Fig. 4.14 Schematic design of the CUS apparatus

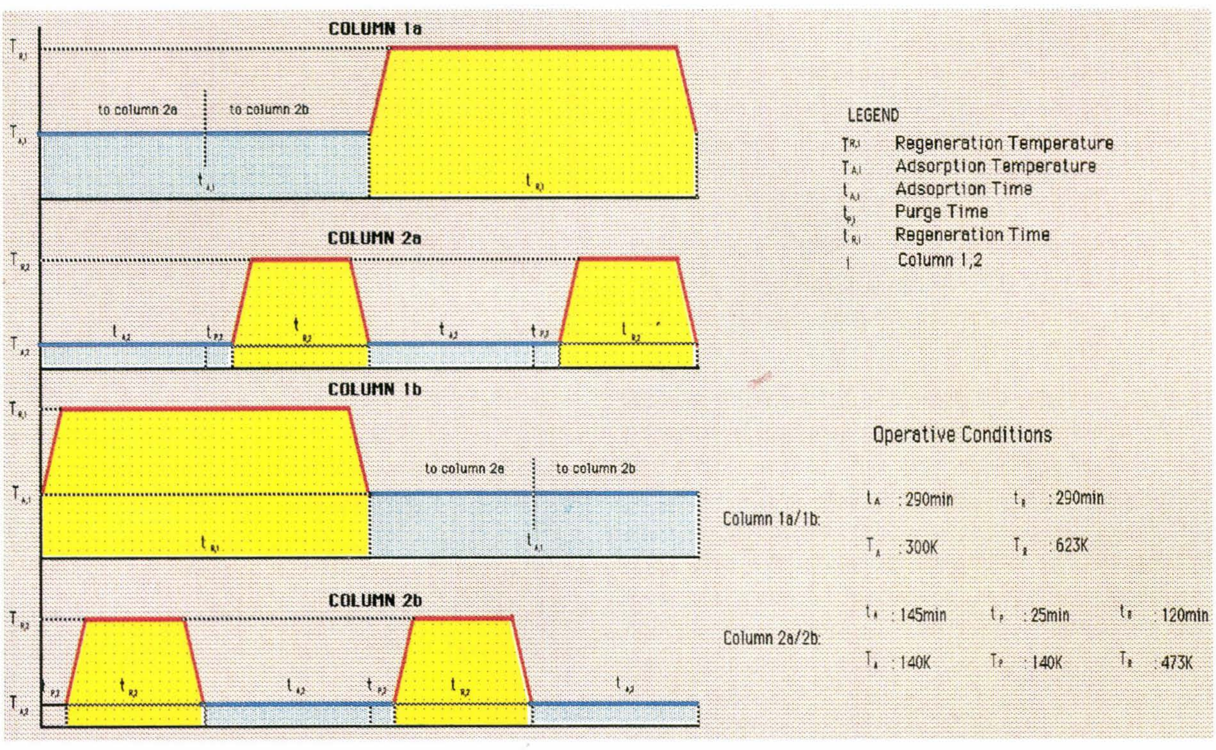

Fig. 4.15 Column operating schedule for the CUS apparatus

to commence within 15 minutes of the start of heating, avoiding significant annealing of the irradiation-induced defects.

The effect of irradiation is to increase the yield stress close to the ultimate tensile strength of the unirradiated material, and to eliminate strain hardening except for the highest test temperatures.

Whereas unirradiated weld material had constantly higher yield stress than the parent plate, after irradiation some weld samples showed a lower yield stress than the irradiated parent plate [15]. This implies that irradiated components may fail preferentially at the weld, although not before significant plastic elongation.

SMA welds have a wider heat-affected zone than the EB welds, but otherwise there was no significant difference between the two types of weld. Some weld material shows a significant, but not disasterous reduction in fracture toughness, as indicated by the degree of necking in the tensile tests.

1.4.5 EUROPEAN TRITIUM HANDLING EXPERIMENTAL LABORATORY (ETHEL)

Status of facility realisation

The progress of ETHEL during 1992 has fallen significantly short of what was anticipated, the main reason being a lack of on-site resources provided by the architect-engineer and a large number of minor technical faults which have hindered component/system testing. Notwithstanding these delays, the installation of the plant is complete while individual system testing is outstanding only for the Heating & Ventilation System. This has allowed the granting of provisional acceptance to the architect-engineer. Nevertheless, various technical discrepancies from the approved revision of the safety report remain to be resolved with the Regulatory Authority. Clearing these non-conformities will generally entail some form of documented justification for changes to system specifications although, occasionally, a plant modification will be unavoidable.

It has been agreed with the Regulatory Authority that, following provisional system acceptance, JRC-Ispra can continue testing systems where no or very minor technical problems are known. Hence, overall system testing (commonly referred to as "prove di sistemi") has commenced and is expected to extend until middle of 1993. During this period, the Regulatory Authority will maintain its surveillance role.

The various procedures for performing these tests of ETHEL systems are being re-drafted to reflect the new responsibility of the Institute. Originally, the architect-engineer was meant to execute the test procedures. Moreover, to recover time, translations into Italian of these procedures are being used as the first drafts of the "prove combinate" procedures, the programme of which is already with Regulatory Authority. It is anticipated that the "prove combinate" and "prove nucleari" will require not more than 8 months, assuming that no grave technical problems occur.

Continuing on licensing activities, the production of other documentation such as technical prescriptions, revised safety report and release formulae, is also underway. Unfortunately, this progress has not been reflected in the area of ETHEL's quality documentation whose production, owing to the unavailability of personnel, is significantly behind schedule. In particular, the development of the laboratory's quality programme and procedures for future operational purposes is still at an embryonic stage.

In the anticipation of taking over the laboratory, the preliminary training of ETHEL operators has been arranged with the architect-engineer. It is also envisaged that a more intensive training programme will be undertaken during 1993 and that some of the operators will be sent to TSTA (Los Alamos, US) for in-field experience at a major tritium facility.

Directly associated with ETHEL was the acquirement of a calorimeter for directly measuring tritium in U-getters to be used in the laboratory for tritium distribution purposes. Calorimetry represents a non-intrusive means of determining tritium inventories and is expected to play a major role in ETHEL's tritium accountancy techniques. Initial tests with the device, using electronic sources and Pu standards, suggest that it can accurately determine tritium quantities in the range 100 - 1500 mW (≅ 0.3-5.0 g tritium) while the sensitivity of the instrument is ≤ 2 mW. Significant interest on the calorimeter has been received from Canada, Germany, Japan and US in the light of increased tritium controls (see below) and it is likely that the calorimeter will be shipped to USA in 1993 for performance testing using actual tritium samples.

Also related to ETHEL was the second JRC Ispra - KfK workshop at which fusion research was presented being conducted in ETHEL and associated Institute hydrogen-deuterium laboratories and in its German sister facility, i.e. Tritium Laboratory Karlsruhe (TLK). The third workshop will be held at Karlsruhe in May 1993.

Finally, given the future importance of ETHEL in tritium research, the selection committee for the Fifth Topical Meeting on Tritium Technology in Fission, Fusion and Isotopic Applications has asked the STI to host the international conference following requests from Ispra, Tokyo and Augusta. The conference is presently scheduled for May 1995.

Tritium control

In parallel with the amendment to the agreement between EURATOM and Canada pertaining to co-operation in the peaceful uses of atomic energy, the "Direction Contrôle de Sécurité (DCS)" in Luxembourg was requested to develop and apply appropriate recording, accounting and inventory techniques for civil European Community tritium facilities. As a result of this commitment, a series of meetings was held in 1991 between DCS representatives and tritium facility operators to establish a basic policy of tritium control. The outcome of these preliminary encounters was that the control of tritium should be based on the EURATOM treaty regulations for safeguarding nuclear materials taking into account the particular properties and characteristics of tritium. Moreover, the tritium facilities were invited to develop a practical tritium control approach derived from operating experience and technical limitations. On this basis, a joint task force was created between personnel from the STI and the Kernforschungszentrum Karlsruhe GmbH (KfK) to develop a translation of the accepted policy into a common tritium control methodology for two civil tritium facilities, namely ETHEL and TLK.

Throughout 1992, this task force has focused on a number of key topics pertinent to tritium control including physico-chemical characteristics of tritium, a typical tritium laboratory, the probable dispersion of tritium in a laboratory and its experimental plants, the envisaged quantification and related uncertainties of tritium in ETHEL and TLK and, finally, a practical approach to tritium control in the laboratories. The work has culminated in a common draft proposal which has been forwarded to the DCS as the basis of detailed negotiations prior to active operations in the laboratories.

Tritium experiments in ETHEL

ETHEL-001:Tritium outgassing from first wall material under fusion reactor conditions

This is the tritium compatible version of the cold equipment dedicated to the study of the recycling of hydrogen isotopes from first wall materials of fusion reactors.

The primary design philosophy for the installation is the effective containment of the tritium needed for the experiments and the minimization of maintenance requirements for tritium contaminated components.

For the containment of tritium, a multilevel containment scheme has been adopted in ETHEL. The first level is made up of the experimental components (i.e. ultra-high vacuum (UHV) experimental chamber, etc.). The second level is provided by a glove box suite of approximately 7 m3 placed in the Laboratory for Exploratory Research (LER) of ETHEL (see ***Fig. A8*** in ref. [1]). Finally, the tertiary containment is provided by the ETHEL laboratory itself (see Appendix A3 in ref. [1]).

The restrictions imposed by having an installation for

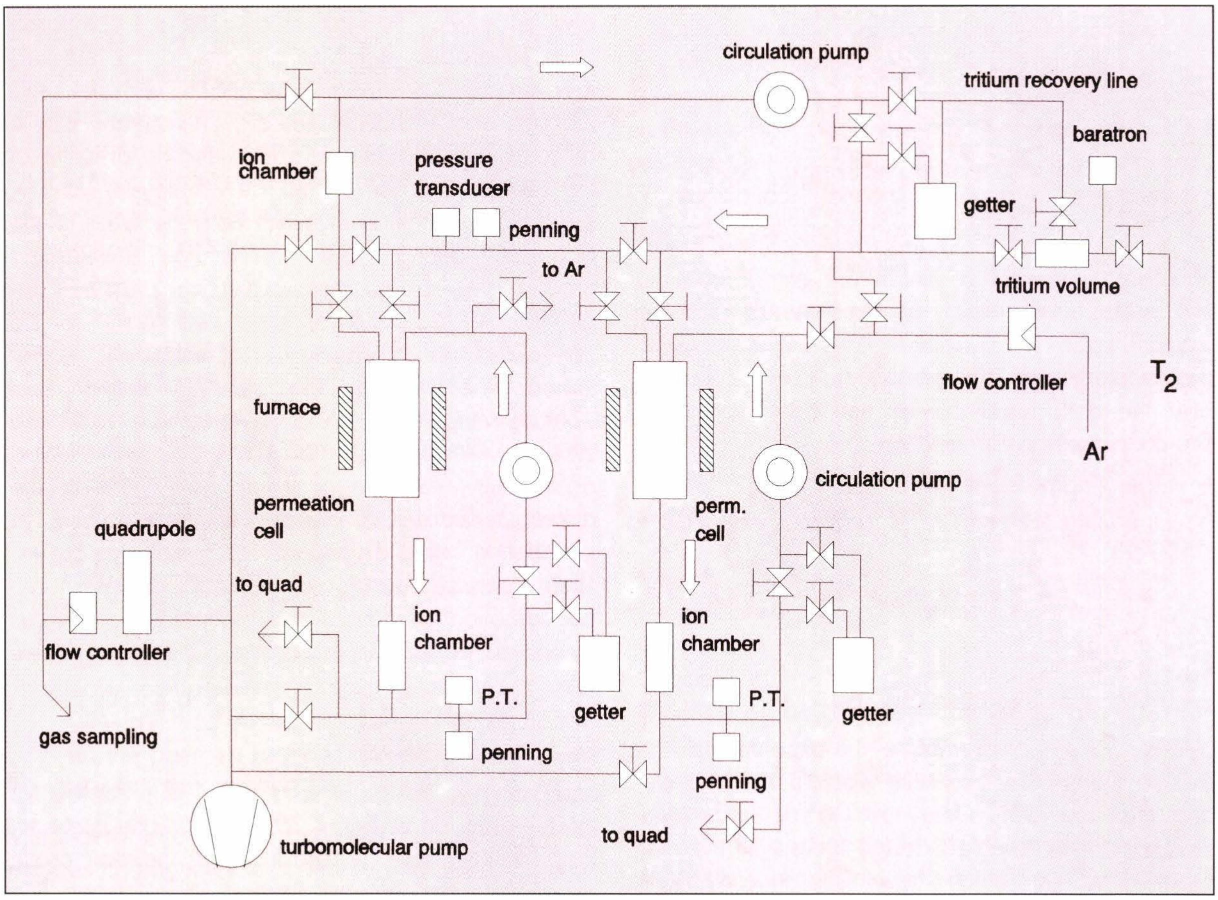

Fig. 4.20 Schematic view of ETHEL-004

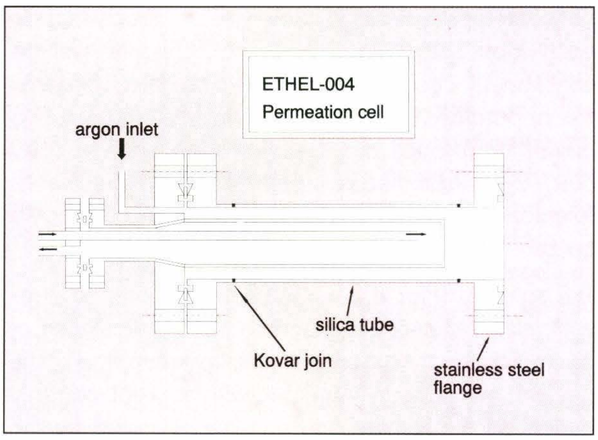

Fig. 4.21 The permeation cell including sample and sample holder

be to prepare the sample. This means welding or otherwise connecting the specimen tube to CF flanges with the appropriate connections. The sample will then be attached to the silica tube and the whole assembly inserted into the apparatus with the furnace in place. Pumping down of the apparatus will then be carried out with a bake-out procedure to reach a good vacuum. When high vacuum conditions have been reached and leak-tightness confirmed, heating of the sample will commence, the temperature of the sample being set at the required value for the permeation experiment. Once the temperature has reached a stable level, the valves allowing the gas to enter the permeation cells will be closed and tritium will be injected into the upstream loop. Next, argon gas will be introduced into the high and low pressure sides of the permeation cell at 1 bar pressure and the circulating pumps started. The final step will be the opening of the permeation cell inlet valves allowing the tritium/argon gas mixture to come into contact with one side of the sample tube. At the same time the monitoring of tritium level on the low pressure side will commence.

The measurement will continue until steady state conditions are reached (i.e. constant permeation rate). Subsequently, a measurement at a different temperature or pressure could be carried out. If the time dependent permeation behaviour is being determined, the high and low pressure sides have to be evacuated before each measurement. The tritium can be absorbed by the flow-through getters attached to each loop, with the pumps ensuring the continuous gas circulation through the getters. The detritiated argon can then be pumped away by the turbopump system.

The important parameters which will need to be recorded continuously during each experiment are: the high pressure of tritium, the pressure of tritium on the low pressure side and the temperature of the sample. It will also be useful to record the ambient temperature and pressure of argon in the apparatus. By the use of suitable interfaces, control and data logging can be effected remotely by computer.

References

[1] GIANCARLI L., BARBIER F., FLAMENT T., FUTTERER M., LEROY P., PROUST E., SANNIER J., RAEPSAET X., TERLAIN A., COEN V., PERUJO A., SAMPLE T., AGOSTINI P., BENAMATI G. - European Research and Development Programme for Water-Cooled Lithium-Lead Blankets, Present Status and Future Work - Presented at the 10th Topical Meeting on the Technology of Fusion Energy, Boston, Massachusetts, USA, June 7-12, 1992

[2] PERUJO A., CAMPOSILVAN J., REITER F., TOMINETTI S. - ETHEL-001: A Versatile Facility for Plasma-Surface Interaction Research - Presented at IAEA Technical Committee Meeting on: 'Atomic and Molecular Data for Fusion Reactor Technology' - Cadarache (France), Oct. 12-16, 1992

[3] PERUJO A., ALBERICI S., CAMPOSILVAN J., REITER F. - Hydrogen in the Martensitic Steel Manet: Diffusivity and Solubility Measurements - Presented at the Fourth Topical Meeting on Tritium Technology in Fission, Fusion and Isotopic Applications - Albuquerque, New Mexico, 29 Sep.- 4 Oct., 1991. Fus. Technol. 21 (1992), pp. 800-805

[4] REITER F., ALBERICI S., CAMPOSILVAN J., SERRA E., FORCEY K.S., PERUJO A. - Diffusivity and Solubility of Hydrogen Isotopes in the Martensitic Steel DIN 1.4914 (MANET) after thermal exposure at 900 K - Presented at the Int. Symposium on Metal Hydrogen Systems, Upsala (Sweden), Jun. 8 -12, 1992, to be published in Z. f. Physik. Chemie

[5] FORCEY K.S., PERUJO A., REITER F., LOLLI-CERONI P. - The Formation of Tritium Permeation Barriers by CVD - Presented at "12th Int. Vacuum Congress/8th Int. Congress on Surface Science" - The Hague, The Netherlands, 12-16 October, 1992, proceedings will be published in J. Nuclear Materials

[6] PERUJO A., DOUGLAS K., AGOSTINI P., CALDWELL-NICHOLS C.J. - Tritium Permeation through Engineering Components: Jet Bellows Experiments - Presented at the "17th Symposium on Fusion Technology", Rome, Italy 14-18 September, 1992

[7] RIEHM M.P., SMELTZER W.W., THOMPSON D.A. - CFFTP Report No. G-86041, 1986

[8] ZARCHY A.S., AXTMANN R.C. - J. Nucl. Mater. 79(1979), pp. 110-117

[9] MALARA C., MENCARELLI T., RICAPITO I., TOCI F., VANSANT E.F. - Characterization of Porous Solids by Analysis of Gas-Physisorption Measurements - presented at the 9th International Zeolite Conference, Montreal, Canada, July 5-10, 1992, (accepted for publication on Microporous Materials)

[10] RICAPITO I., MALARA C., PIERINI G., FACCHINI A., MENCARELLI T., SPELTA B., TOCI F., VIOLA A. - The Adsorption of Gas Mixtures in Zeolites: Experimental Determination and Theoretical Prediction of Equilibria - 17th SOFT, Roma, 1992

[11] MALARA C., PIERINI G., VIOLA A. - Correlation, Analysis and Prediction of Adsorption Equilibria - EUR 13996 EN, 1991

[12] PIERINI G., SPELTA B., MALARA C., RICAPITO I. - Engineering of Gas-Solid Processes: Hydrogen Isototpes Separation - presented at the 2nd ETHEL/TLK Workshop on Safety Technology in Thermonuclear Fusion, Ispra, Varese, Italy, 6-7 Feb. 1992

[13] MALARA C., MENCARELLI T., RICAPITO I., TOCI F. - Characterisation of Substrates and Evaluation of Adsorption Parameters - presented at the 2nd ETHEL/TLK Workshop on Safety Technolgoy in Thermonuclear Fusion, Ispra, Varese, Italy, 6-7 Feb. 1992

[14] MALARA C., RICAPITO I., SPELTA B., TOCI F., VIOLA A. - Adsorptive Removal of the Impurities from the Exhausted Plasma - Presented to the 9th International Zeolite Conference, Montreal, Canada, July 5-10, 1992, (accepted for publication on Microporous Materials)

[15] EDWARDS R.A.H., ELEN J.D., BOTTELIER P., FACELLI R. - FRUST Project Test Report: Tensile Testing of Irradiated Samples from Welds in Solution Annealed 316L Austenitic Steel Plate - Technical Note I. 92.114, Nov. 1992

1.5 INDUSTRIAL HAZARDS

The general objective of the work is the assessment, improvement and harmonization of safety methodologies, focusing on the key areas:

- *control of chemical reactions with a potential for thermal runaway,*
- *emergency pressure relief of batch chemical reactors and storage vessels,*
- *prediction of the dispersion of dense vapour clouds,*
- *gas cloud explosion hazards.*

1.5.1 BATCH CHEMICAL REACTORS

Research on process dynamics of discontinuous chemical reactors, in conditions close to runaway, is carried out in the FIRES Project (Facility for Investigating Runaway Events Safely). The main objective of FIRES is the study of off-normal behaviour in batch and semi-batch processes in a scale closer to industrial practice than calorimetric laboratory experiments. Specifically, this study intends to:

- check and develop criteria for the safety of processes by studying the characteristics of reactive mixtures and determining the critical operating conditions,
- test and develop measures for the prevention of uncontrolled thermal excursions and the associated over-pressurisation phenomena, including control and early detection systems, and interlocks,
- apply the knowledge gained for developing an intelligent tool to assist in the design and optimisation of such processes.

The FIRES project is divided in three different parts:

- A calorimetric laboratory which is used to gain basic knowledge of the chemical processes prior to their investigation in the FIRES reactor. The laboratory has been equiped with a differential scanning micro-calorimeter, a small scale reactor and an adiabatic calorimeter for venting studies. Furthermore, Gas Chromatography (GC), infra-red (IR) and ultraviolet (UV) spectroscopy are used for chemical analisis.
- A mathematical simulator of FIRES (FISIM, FIres SIMulator) that serves as an aid and a complement to experiment design, and subsequent data analysis [1]. Furthermore, a special version of the simulator is applied on-line for the estimation of non-measured variables and main parameters of the process, such as kinetics and heat transfer characteristics [2].
- A fully automated 100 l pilot plant reactor installed within a bunker (***Fig. 5.1***), equiped with sensitive measuring devices, and provided with an early warning detection system, shut-down systems, and emergency pressure relief, so that hazardous chemical reactions can be investigated safely [3].

The chemical processes, on which attention is presently focused, are the aromatic nitrations by mixed acid. The main reason for choosing aromatic nitrations was that, despite the fact that they are one of the oldest and most common industrial reactions, there are still a considerable number of problems in predicting their behaviour in discontinuous reactors. This is due to the fact that nitrations involve simultaneously chemical reaction and mass transfer phenomena, which leads to complex phenomena while characterizing and scaling-up these processes. Compared with homogeneous systems, the analysis of heterogeneous systems is further complicated since it has to include other aspects such as distribution coefficients between the phases, droplet sizes, phase inversion etc., which may affect the reaction rate. In addition, nitrations involve high exothermicity and side reactions, and they have produced a considerable amount of incidents, provoked by runaway reactions [4]. In particular, toluene mononitration by mixed acid was taken as a specific case due to the amount of literature available [5-7].

Fig. 5.1 The FIRES project facility

Small scale studies

The mononitration of toluene has been taken as a specific case and an extensive experimental programme has been carried out using adiabatic and heat flow calorimetry, supported by chemical analysis [8]. After the preparatory experiments [9] and the experiments on normal industrial conditions [10], the research continued on the assessment of methods to stop the runaway and on the effect of phase inversion:

- *In-situ containment of a thermal runaway during toluene nitration* [11]: The nitration kinetics is strongly dependent on the sulphuric acid strength which can be expressed to be:

$$H_2SO_4 \text{ strength} = \frac{\text{mass of sulphuric acid}}{\text{total mass of water + sulphuric acid}}$$

Consequently, the sudden addition of water to the reaction mass will decrease this strength and hence the overall reaction rate. The main problem comes from the fact that an appreciable amount of heat due to dilution of water in mixed acid will be produced at the same speed at which the water is introduced in the reactor (instantaneous reaction). To check the possibilities to use the fast injection system of the FIRES facility, in case of emergency, different experiments were carried out with the reaction calorimeter RC1. In these batch experiments an uncontrolled thermal excursion was produced by adding batchwise 30% molar of HNO_3 in relation to toluene. *Fig. 5.2* shows the results of the reaction carried out without and with the addition of water: in the second case the reaction is completely stopped. The sudden increase in temperature is due to heat of dilution. After this study, it was demonstrated that the fast addition of water is a better protection system than the venting or dumping of reactor contents, since no product is lost and the reaction can be restarted again by adding enough concentrated sulphuric acid.

- *The effect of phase inversion during toluene nitration* [12]:
  The accumulation of unreacted mixed acid in semibatch aromatic nitrations can be dangerous if accompanied by phase inversion, due to the fact that the reaction rate can increase suddenly since, in industrial operating conditions, it is mass transfer controlled. *Fig. 5.3* shows the comparison between experimental and simulated

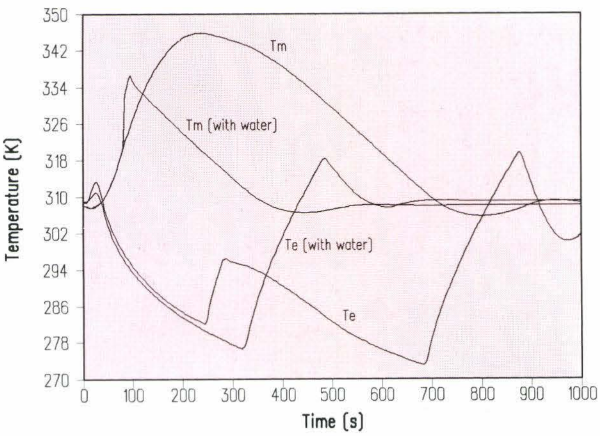

Fig. 5.2 Experimental reactor temperatures in RC1 batch experiment. Addition of 285 g of mixed acid (comp.: 56.2 wt.% H_2SO_4 and 28.8 wt.% HNO_3) to 600 g of toluene. Reactor temperature $T_{m_{set\text{-}point}}$ = 308 K and stirrer speed = 6.67 s^{-1}.
In the second experiment, addition of 100 g of water was carried out when the reactor temperature reached approx. 318 K.

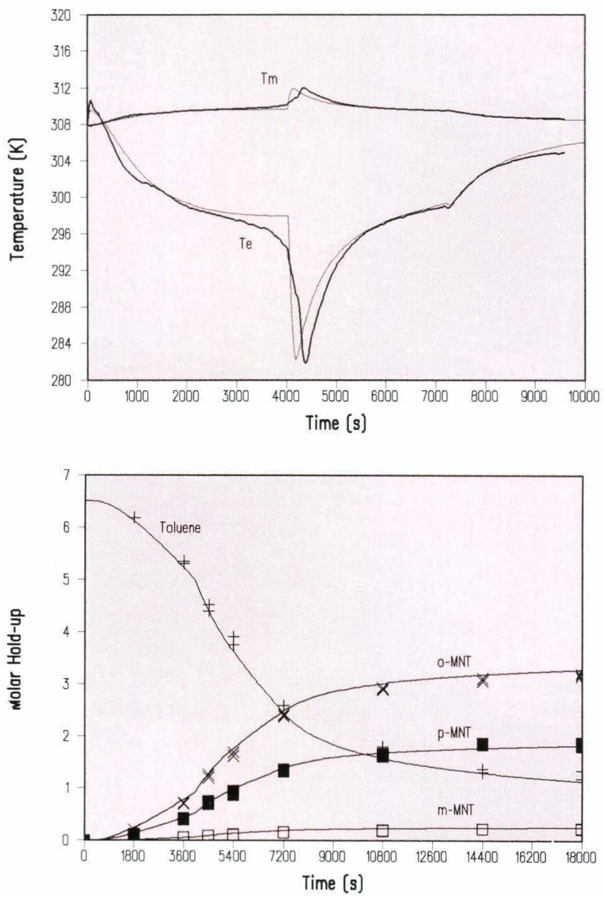

Fig. 5.3 Experimental and simulated temperatures and molar hold-up of the reactants in a semibatch nitration. 1570 g of mixed acid (68.0 wt.% H_2SO_4 strength) were added to 600 g of toluene in 2 h. $T_{m_{set\text{-}point}}$ = 308 K and stirrer speed = 6.67 s^{-1}.

reactor and jacket temperature profiles versus time. The agreement between the curves shows that the effect of phase inversion could be successfully simulated.

- *Other systems:*
  The study of esterification reaction between 2-butanol and proprionic anhydride has been continued [13,14]. Data from these experiments are used to validate scaling criteria for heat transfer and runaway prediction, as well as the different methods for kinetic investigation.

Pilot reactor scale studies in FIRES

The FIRES facility has become fully operational. The experimental programme carried out comprised:

- **Mixed acid preparation:** Different mixed acids (H_2O-HNO_3-H_2SO_4) were prepared to be used as reagent during nitration experiments. Furthermore, these experiments served as an evaluation of the possibilies to use FIRES reactor as a calorimeter. ***Fig. 5.4*** shows the comparison between the experimental and simulated [9] heat flow due to dilution. The agreement obtained showed that the FIRES facility can be used as a reaction calorimeter, and as a consequence the overall reaction rate can be obtained for any process studied.
- **Normal industrial conditions:** Different experiments were carried out, modifying the operating conditions (temperature, feeding rate, stirrer speed, and sulphuric acid strength) to understand their influence. ***Fig. 5.5*** shows the comparison between the experimental and simulated molar hold-up for a semibatch nitration.

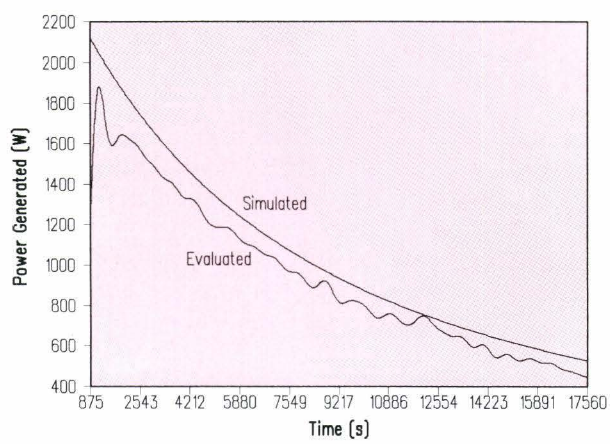

Fig. 5.4 Evaluated and simulated rates of heat generation during the preparation of mixed acid in the FIRES reactor. 91 kg of H_2SO_4 (96 wt.%) added to 52 kg of HNO_3 (65 wt.%) in 4.64 h.

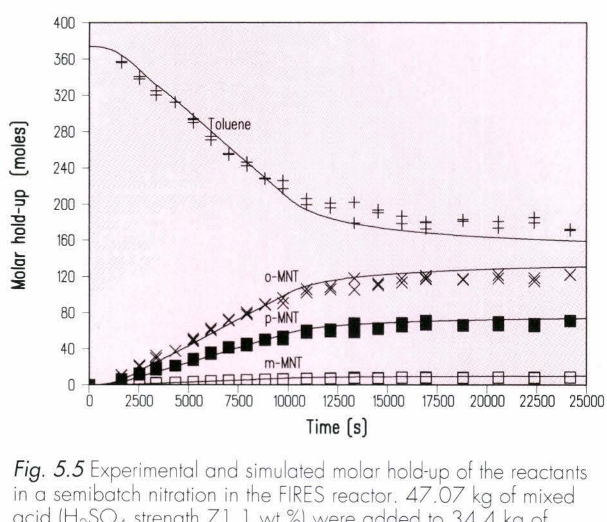

Fig. 5.5 Experimental and simulated molar hold-up of the reactants in a semibatch nitration in the FIRES reactor. 47.07 kg of mixed acid (H_2SO_4 strength 71.1 wt.%) were added to 34.4 kg of toluene in 3 h. $T_{m_{set\text{-}point}}$ = 308 K and stirrer speed = 4.92 s^{-1}.

- **Simulation of failures:** Agitation is an important parameter which sometimes is underestimated in its potential to create hazardous situations. In fact, in an analysis of incidents in industrial batch reactors [4], it was found that 10-13% of incidents were due to problems with the stirrer (i.e. loss of agitation, insufficient mixing). In a preliminary experimental analysis, a Plexiglas reactor (scale 1:1) was used to determine qualitatively the stirrer influence. In this reactor, different tests were performed adding water in which plastic pellets (p= approx. 1.3 g/cm^3) were present. It was observed that there was no homogenisation below a certain volume (approx. 38 l). The denser phase (the pellets) remained mainly at the bottom of the reactor. In view of that, three nitration experiments (***Fig. 5.6***) were carried out to simulate these stirring problems for homogenising the reacting mass below a certain volume. As can be seen, the temperature reactor behaviour is completely different from one experiment to the other, even though the only parameter changed was the initial volume of toluene present in the reactor. Furthermore, as the reactor volume reached the critical value, all the accumulated mixed acid at the bottom of the reactor started to react producing a thermal excursion.
- **Stopping the runaway by fast injection of a reaction suppressant:** In the runaway experiment of ***Fig. 5.6***, when the reactor temperature reached 60°C, the fast injection system was operated: 18 l of water were introduced into the reactor and the reaction was stopped. The maximum temperature and power generated were 72°C and 120 kW respectively. After the reaction, the organic phase was analysed by GC and less than 0.06% of 2,6 dinitrotoluene was found, which implies that the heat generated was only due to mononitration and dilution.

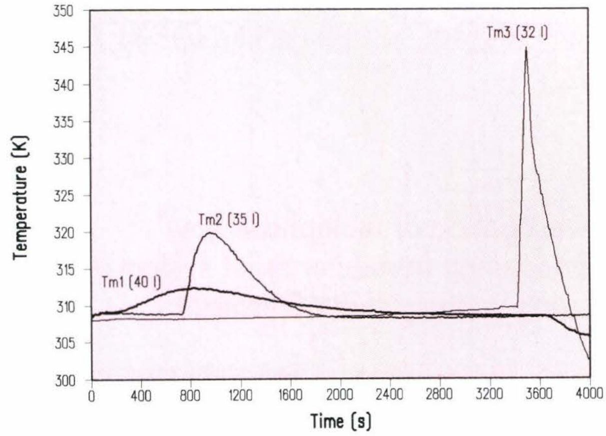

Fig. 5.6 Experimental reactor temperatures for three semibatch nitration experiments. Mixed acid of 80% H_2SO_4 strength added to toluene at 2.44 g/s, $T_{m_{set\text{-}point}}$ = 308 K and stirrer speed = 4.92 s^{-1}.

1/initial volume of toluene = 40 l

2/initial volume of toluene = 35 l

3/initial volume of toluene = 32 l

Simulation of batch chemical processes (FISIM)

A version of FISIM for the FIRES reactor was developed and experimentally validated. The model for predicting the overall nitration rate was introduced in FISIM, and from the comparison between the experimental and simulated results it was shown that the model was able to predict accurately the dynamic behaviour of RC1 toluene mononitration experiments and the influence of the main operating conditions [2, 9-12]. This model has the scaling-up correlations and was consequently used without modifications to predict the overall reaction rate of toluene nitration carried out in the FIRES reactor. The agreement obtained (***Fig. 5.5***) was satisfactory and demonstrated the validity of the numerical simulation to understand the behaviour of nitration plants and its usefulness to carry out complex scaling-up procedures, since not only heat transfer but also mass transfer were involved.

the design of relief systems for packed-bed reactors containing foamy fluids since the relief system requires to be of sufficient size to accommodate also the discharging spheres. Also, possible accumulation points for the spheres should be avoided (safety valves should not be used).

Two new venting facilities are now being constructed. The design, ordering and delivery of the vessels and pipework were completed and both facilities are expected to be operational in 1993. One of the facilities, known as COLUMBUS, will permit studies of the venting characteristics of long horizontal vessels while the other, known as DRACULA (Depressurization, Relief And Containment Using Large Apparatus) is intended for venting studies at industrial-scale in order to validate (or otherwise) design methodologies developed on the basis of data from small-scale experiments. The above facilities are shown in schematic form in ***Figs. 5.8*** and ***5.9***.

Nuclear Magnetic Resonance (NMR) mass flow measurements

The NMR mass flow measurement [22-23] has been improved by a further development of the evaluation software. Measurements have been carried out at our oil-water-gas loop with water and with water-nitrogen mixtures. ***Fig. 5.10*** shows a calibration curve obtained with water only. Plotted is the average velocity V_{NMR} measured by NMR versus the same velocity V obtained from a turbine in the loop. The accuracy of the velocity V_{NMR} lies well within ±2%.

Fig. 5.11 shows the same velocity in the case of two-phase water-nitrogen flow. The velocity V has been determined from water flow, nitrogen flow, pressures and temperatures measured in the loop. V_{NMR} has been measured as above. The scatter in this velocity is due to fluctuations in the two-phase flow, which do not show up in the loop velocity V. The accuracy of V_{NMR} can thus be given a value of ±5%. It has to be emphasized that this is not the accuracy of the NMR measurement in itself, it is only that the fluctuations are also measured by NMR because the two-phase flow does not exhibit exactly pseudostationary character.

In ***Fig. 5.12*** shows the liquid fraction ε_{NMR} obtained by NMR versus the same fraction ε obtained from loop data. The scatter in ε_{NMR} is again due to fluctuations in the two-phase flow which are measured by NMR as in the case of ***Fig. 5.11***. The accuracy is therefore also about ±5%.

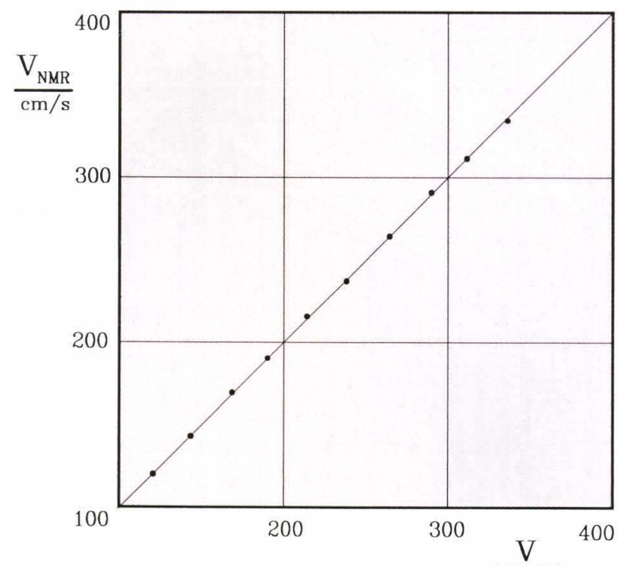

Fig. 5.10 Velocity V_{NMR} measured by NMR in single phase water flow vs. velocity obtained by turbine

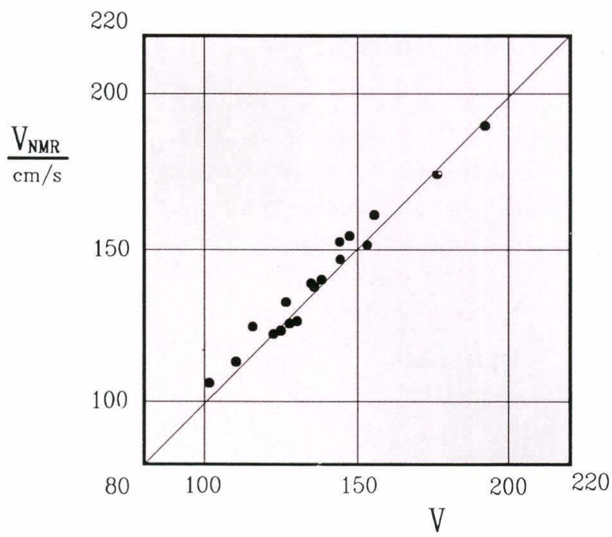

Fig. 5.11 Velocity V_{NMR} measured by NMR in two-phase water-nitrogen flow vs. velocity obtained parameters of single phases in the loop

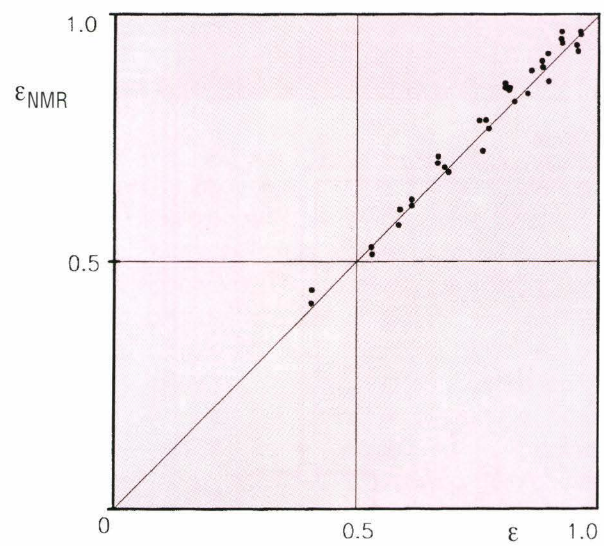

Fig. 5.12 Liquid fraction ε_{NMR} obtained by NMR in two-phase water-nitrogen flow vs. ε obtained by parameters of the single phases in the loop

The emergency relief system code RELIEF [24,25]

RELIEF is a transient, one-dimensional, multicomponent, two-phase fluid dynamic computer code that simulates the behaviour of chemical batch reactors and storage vessels prior to and during a thermal runaway event. An arbitrary number of chemical species (both inert and reactive) are modelled and the chemical conversion, the mass transfer between the component liquid and vapour phases, the two-phase fluid dynamics and the interactions among these processes are described. The model is coupled to a sophisticated input and output processing facility (developed at the JRC) and thereby creating a complete self-contained package. The main aim of this work is to provide chemical industry, technical institutions and public authorities with a validated computer code that will help create harmonised rules for the design of emergency pressure relief systems at a European level.

During 1992 much effort has been devoted to the completion of the "stand alone" version of RELIEF. Every effort has been made to minimise the interaction between the user and the code, which enables a non-specialist to set up a problem and quickly obtain realistic results. A comprehensive on-line help facility gives the user guidance and recommendations to further ease the task for input of data. There are also consistency, limit and syntax checks so that unrealistic data cannot be entered, and a pictorial mimic of the input data greatly aids error detection. A variety of vessel shapes and discretisation schemes can be selected with the minimum of input data being requested; similarly wall thermal capacity and heat transfer can be modelled. The results of the calculation are displayed graphically (with the option of a postscript hard copy), and an additional animation feature allows the user to visualise the transient in faster than real time. Examples of the screen layout and the simulation option are shown in ***Figs. 5.13*** and ***5.14***.

An underlying feature of emergency pressure relief design is often the lack of data regarding component physical properties and chemical kinetics. For this reason possible predictive features have been incorporated into RELIEF. These include vapour pressure behaviour, surface tension, non-ideal liquid and vapour specific volume behaviour as the critical region is approached. A user generated physical property database is included in the system, which means that once the component data has been entered into the database the user, at the time of problem specification, needs only to enter the component name or its Chemical Abstract Registry Number (CARN) to retrieve all the relevant physical property informations.

Due to the high computational speed of RELIEF parametric studies can easily be performed making the package ideal for design purposes. Similarly, the possibility of modelling multiple vent locations with independent safety valve set pressures enables the advantages of combined top and bottom venting over top venting to be studied quantitatively.

The package RELIEF is now operational on a workstation and it is planned to extend it to a PC platform in the near future.

In addition to the development of RELIEF, the methodologies contained in this code are being further developed in order to simulate the transient behaviour during runaway of a "catalyst bed reactor" operated at supercritical conditions. The analysis undertaken requires a special treatment of the fluid expansion and chemical conversion processes. The chemical conversion is modelled taking into consideration the combined effects of chemical reaction and diffusion in the pores of the catalyst pellets and of diffusion in the fluid region outside the pellets.

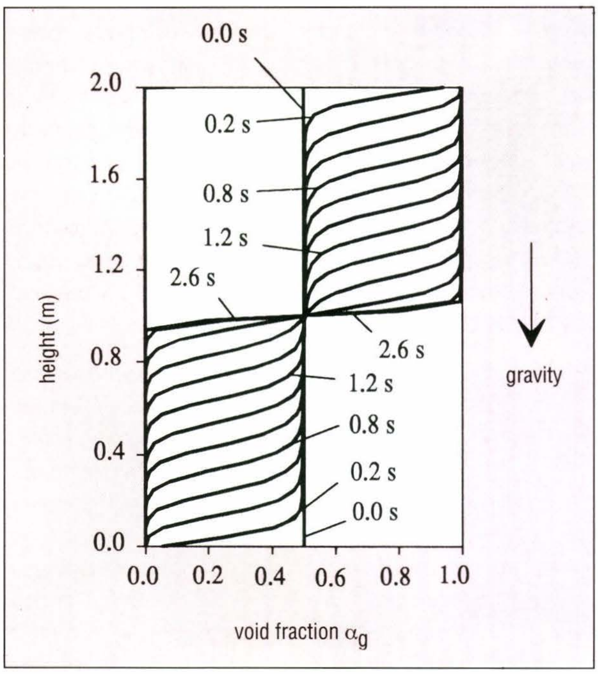

Fig. 5.15 Sedimentation problem, void distribution at different time values

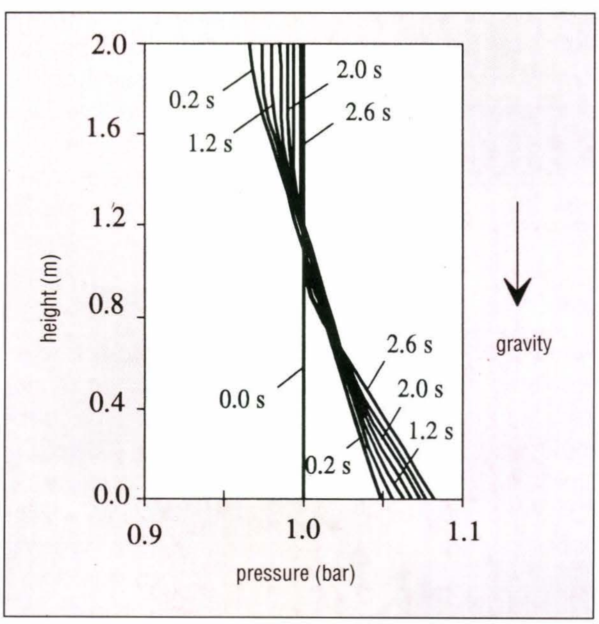

Fig. 5.16 Sedimentation problem, pressure distribution at different time values

controlled by two-phase discharge from the pipe and the continuous evaporation (flashing) of the liquid.

A comparison between measured and predicted values is shown in *Figs. 5.17* and *5.18* for the pressure at the pipe head and the void fraction at the mid section of the pipe. The results show a generally good agreement, apart from an anomaly in the measured void fraction at the first period of the transient for which no explanation has been given by the experimentalists.

The work foreseen for 1993 will include the improvement of the description of the interfacial transport processes for mass, momentum and energy by adding an additional balance equation for the interfacial area concentration. This work will be performed in cooperation with the Institute for Thermodynamics of the University of Stuttgart. A further improvement of the numerical method includes an implicit treatment of the pressure wave propagation process and an extension of the numerical scheme to provide second order accuracy in space. However, the major effort will be directed to extend the numerical techniques to handle two-dimensional two-phase flow processes on the basis of unstructured grids.

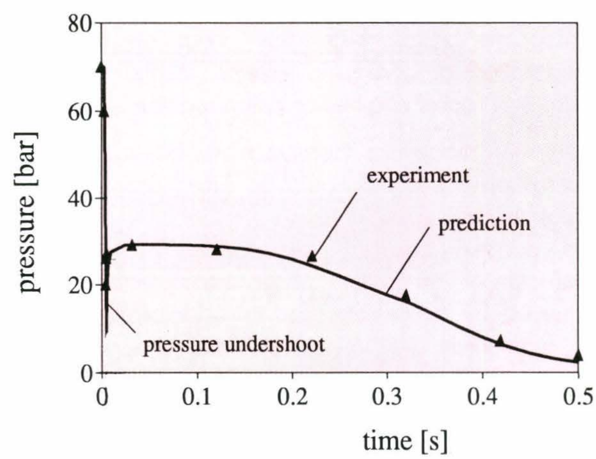

Fig. 5.17 Edward's pipe, pressure at pipe head

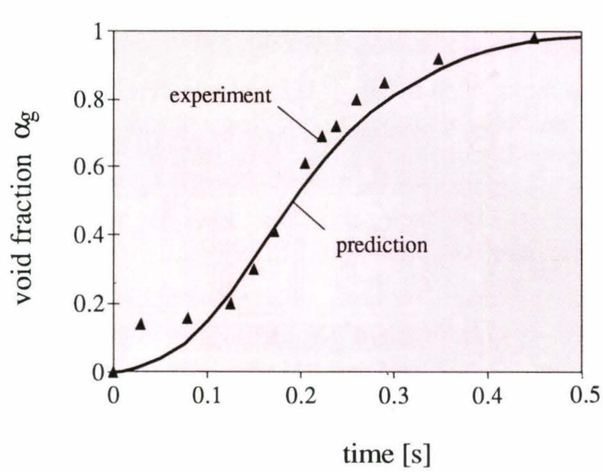

Fig. 5.18 Edward's pipe, void fraction at pipe mid section

1.5.3 DISPERSION OF DENSE VAPOUR CLOUDS

The general objective of this project is to develop and further improve computer models describing dispersion of denser-than-air vapour clouds in irregular terrains with obstacles. The two main applications are assessment of accident consequences and optimization of safety engineering features for accident mitigation. The work consists of validation and further development of the three-dimensional finite volume code ADREA-HF and simpler shallow layer models. The research is carried out in the frame of the existing collaboration contract with the Demokritos Research Centre of Greece and in association with the CEC sponsored STEP project on dispersion of two-phase flashing releases (FLADIS).

The ADREA-HF code has been applied to validate its capability to describe dispersion in presence of obstacles. The Thorney Island Trial 21 was chosen for simulation. This experiment concerned the instantaneous release of 2000 m^3 dense gas at the centre of a semi-circular fence with height 5 m and diameter 50 m. A special technique was adopted in order to model the fence without having to use an excessively fine numerical grid [30]. The effects of the fence on the dispersion of the gas are predicted successfully (*Fig. 5.19*) and the results appear to be in satisfactory agreement with the experimental data (*Fig. 5.20*). Using the experimental data of the Thorney Island Trial 8 and the Desert Tortoise Ammonia Experiment 1, ADREA-HF has been applied to validate and further develop the turbulence models contained in the code [31], [32] and [33]. A transport equation for the calculation of concentration fluctuations has been implemented in the code. This should allow to evaluate the uncertainty in the mean concentration value, by taking into account the statistical nature of the turbulent dispersion. The task of quantifying the terms in this equation has not yet been completed.

The 1-D shallow layer model [34], describing a 3-D cloud by integration over the cloud height and width, has been validated by the application of wind tunnel and field test data. A rather simple relation for the entrainment velocity was shown to give good predictions for the concentration at the ground level for both instantaneous and continuous releases with and without wind. An example of a calculated cloud propagation for an instantaneous release with wind is shown in *Fig. 5.21*. An extension of the model to two dimensions, where variations over the width of the cloud is calculated, has been initiated. Preliminary calculations for unobstructed terrain without entrainment have been performed.

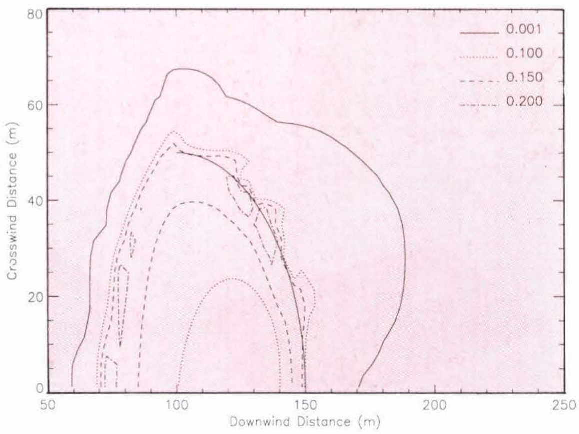

Fig. 5.19 Dispersion of a dense gas cloud in presence of a circular fence (simulation of Thorney Island Experiment No. 21 with ADREA-HF): Contour plot of dense gas mass fraction, at the horizontal (ground) plane, 30 s after the release of the gas

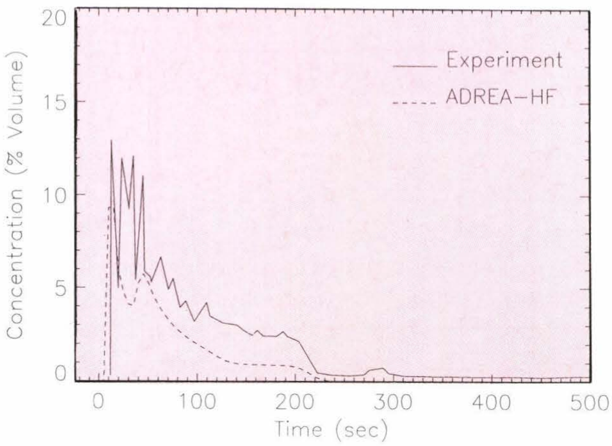

Fig. 5.20 Experimental and calculated dense gas concentration, as a function of time, 40 m from the point of the release (Thorney Island Experiment No. 21)

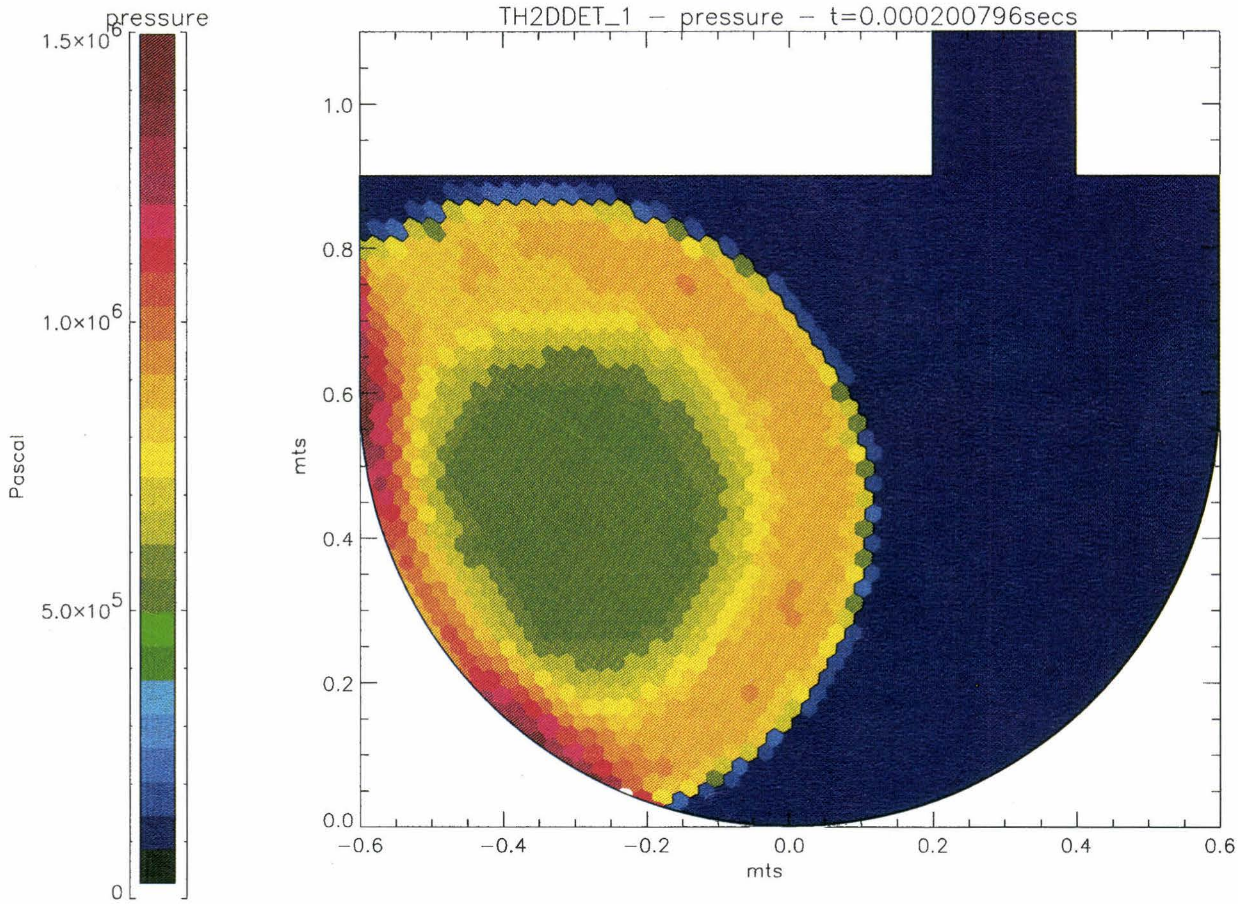

Fig. 5.24 Two-dimensional simulation of a detonation in a closed vessel filled with a mixture of hydrogen and air, stochiometric in hydrogen and oxygen. Situation at time 2×10^{-4}s after ignition

- As a slow combustion develops, the ratio between the gas mixture velocity and the sound speed (Mach number) may remain far below unity.
- Heat, momentum and species transfers by diffusion also bring in different time/length scales, far smaller than the integral flow scale.

For realistic flow simulations, only a narrow range of length/time scales is of practical relevance, to which corresponds an ideal grid mesh size and time step size.The coexistence of much smaller scales, termed as stiffness, causes difficulties such as numerical instabilities and loss of accuracy. These scales, smaller than the grid size, cannot be described by any numerical scheme and have to be modelled in a somewhat integral fashion, in order to preserve global properties such as the normal laminar flame speed,dissipation rate, friction factors, Nusselt numbers.Thermodynamic relationships should also be preserved, at least in average.

Most of the ongoing activities directly address the problem of multiple time/length scales:

- New algorithms have been designed for the simulation of sets of elementary chemical processes. To circumvent the stiffness problem, the model itself has been altered: the fastest chemical reactions have been assumed to reach their equilibrium within a small fraction of the time step. The modified model is formulated as a differential-algebraic system of equations. The differential part consists of non-stiff ordinary differential equations whose unknowns are linear combinations of the species partial densities and accounts for the limiting chemical processes. The algebraic part, established on thermo-dynamics grounds, is formulated as an entropy maximization problem under certain linear constraints. It plays the role of an equation of state which links a reduced number of degrees of freedom. In addition, alternative numerical schemes have been designed for the gas dynamics, taking advantages of the problem size reduction [38].

- Any extension to non-explosive combustion regimes (slow deflagrations, diffusion flames) imposes the use of implicit time advancement schemes to circumvent the too stringent stability limits. A fully implicit finite element-like solver has been implemented to account for the heat, momentum, species diffusion and the heat transfer across the boundaries. Its solution is searched within a separate substep. Its compatibility with the finite-volume Euler equations solver is ensured performing a point-wise conservative reconstruction of the solution from the cell averages, with the constraint to fulfil the boundary conditions of the parabolic problem [39]. This solver has been verified against classical parabolic test problems.

An original semi-implicit scheme [41] has been designed for the solution of the multicomponent Euler equations over a wide range of (subsonic) Mach numbers, including the stiff limit of nearly incompressible flows. The main requirements of this scheme were:

- to circumvent the sonic Courant stability condition,
- to insure the conservativity,
- to retain the time and space second-order accuracy, at least for the convective characteristic fields with an accurate non-oscillatory capturing of the contact discontinuities,
- to advance explicitly in time the species mass conservation equations in order to reduce the size of the discrete non-linear algebraic equations system,
- to include a predictor step able to provide an initial guess for the solution of the non-linear system.

The implementation has been performed on a two-dimensional unstructured grid, and verified against various one-dimensional test problems.

Fig. 5.25 represents the evolution of the density profile along a subsonic shock tube. The contact discontinuity appears accurately captured. The shock and the rarefaction wave, leaving the computational domain,are significantly spread whilst retaining their correct intensity.

The space stiffness problem has been addressed using a grid adaptation methodology, based on the automatic addition or suppression of grid nodes. The flow regions in which a mesh enrichment is expected to improve the accuracy of the calculation correspond in general to high gradients/source terms: reactive layers, shocks, discontinuities, boundary layers, jets and wakes. Such a methodology includes the definition of local criteria for altering the nodes density, an algorithm to update the grid itself and an interpolation of the discrete solution onto the new grid.

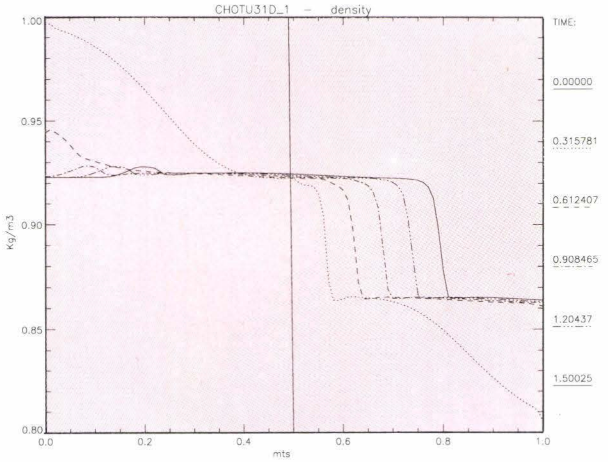

Fig. 5.25 Subsonic shock tube calculation using a semi-implicit time discretization. Density profiles at different times

During the past year, the practicability of the procedure has been firmly established. In order to maintain a tractable system at a first stage, relatively simple techniques have been chosen with respect to local criteria selection, grid processing and interpolation. A window-based data animation system has been developed which displays, together with the calculated transient results, the location of the node points which are being added or removed ([42] and [43]).

The present grid adaptation system has been tested against transient calculations involving single fluid shock reflections. *Fig. 5.26* represents the pressure field history of the Mach reflection of a shock wave on a solid boundary at a given angle of incidence, using the unstructured adaptive grid procedure.

Further developments will concern:

- the definition of local criteria for refinement to be applied to problems involving flame fronts and boundary layers,
- the implementation of more accurate interpolation procedures,
- the coupling with the gas dynamics implicit solvers.

The postprocessing of the results uses the graphic package TURCOM newly developed at ISEI-Ispra for

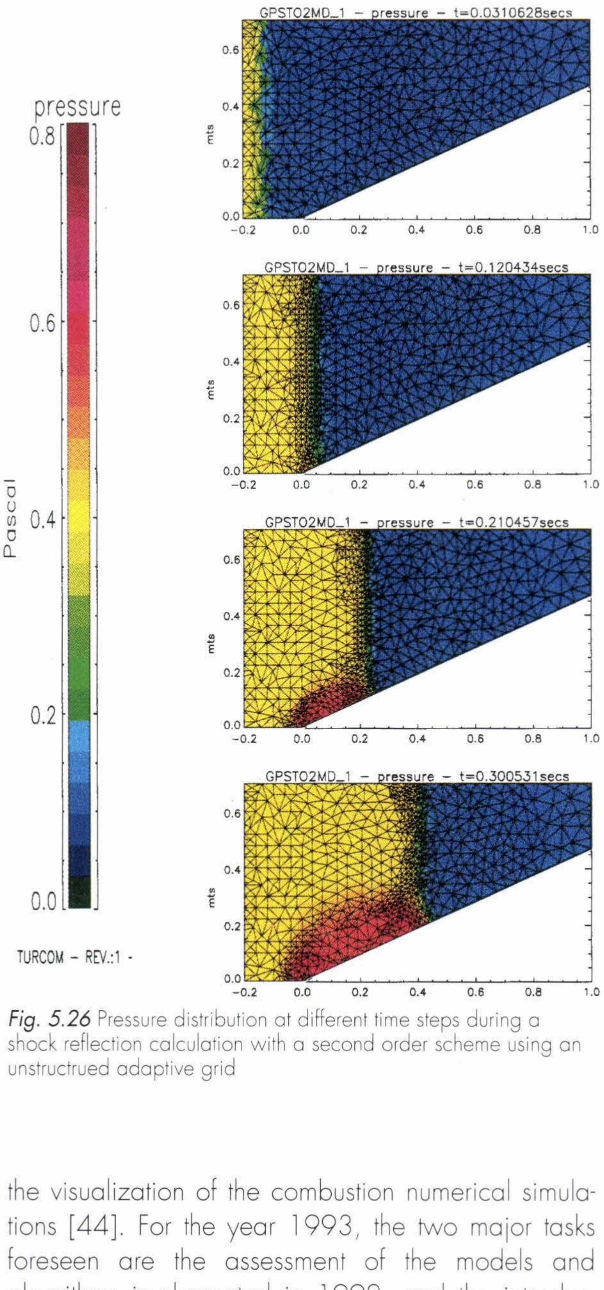

Fig. 5.26 Pressure distribution at different time steps during a shock reflection calculation with a second order scheme using an unstructrued adaptive grid

the visualization of the combustion numerical simulations [44]. For the year 1993, the two major tasks foreseen are the assessment of the models and algorithms implemented in 1992, and the introduction of turbulent combustion models.

References

[1] ZALDÍVAR J.M., HERNÁNDEZ H., BARCONS C. - Development of a mathematical model and numerical simulator for a reaction calorimeter. FISIM, RC1 version - Technical Note No. I.90.109, Commission of the European Communities, JRC Ispra (Italy), 1990

[2] HERNÁNDEZ H., ZALDÍVAR J.M., BARCONS C. - Development of a mathematical model and a numerical simulator for the analysis and optimization of batch reactors - Computer Chem. Engng., 17S, pp. 45-50, 1993

[3] HERNÁNDEZ H., ZALDÍVAR J.M. - The JRC FIRES project for investigations on runaway reactions - Eurotherm Seminar No. 14, Proceedings of Heat Transfer and Major Technological Hazards, 13, 1990

[4] BARTON J.A., NOLAN P.F. - Incidents in the chemical industry due to thermal-runaway chemical reactions - in Safety of Chemical Batch Reactors and Storage Tanks, Kluwer Academic, 1-18, 1991

[5] SCHOFIELD K. - Aromatic Nitration - Cambridge University Press, 1980

[6] OLAH G.A., MALHORTRA R., NARANG S.C. - Nitration: Methods and Mechanisms - VCH Publishers Inc., 1989

[7] ALBRIGHT L.F., HANSON C. eds. - Industrial and Laboratory Nitrations - ACS Symposium Series No. 22, 1975

[8] MOLGA E., BARCONS C., ZALDÍVAR J.M. - Mononitration of toluene and quantitative determination of the isomers distribution by gas chromatography - (accepted for publication in Afinidad), 1993

[9] ZALDÍVAR J.M., HERNÁNDEZ H., BARCONS C., NOMEN R. - Heat effects due to dilution during aromatic nitrations by mixed acid in batch conditions - (accepted for publication in Journal of Thermal Analysis), 1992

[10] ZALDÍVAR J.M., BARCONS C., HERNÁNDEZ H., MOLGA E., SNEE T.J. - Modelling and optimization of semibatch toluene mononitration with mixed acid from performance and safety viewpoints - Chem. Eng. Sci., 47, pp. 2517-2522, 1992

[11] ZALDÍVAR J.M., BARCONS C., HERNÁNDEZ H., MOLGA E. - In situ containment of a thermal runaway during toluene nitration - (submitted to Chemical Engineering and Processing)

[12] ZALDÍVAR J.M., BARCONS C., HERNÁNDEZ H., MOLGA E. - The effect of phase inversion during semibatch toluene mononitration with mixed acid - (submitted to Chemical Engineering Journal)

[13] SNEE T.J., BASSANI C., LIGTHART J.A.M. - Determination of the thermokinetic parameters of an exothermal reaction using isothermal, adiabatic and temperature-programmed calorimetry in conjunction with spectrophotometry - (accepted for publication in Journal of Loss Prevention), 1993

[14] SNEE T.J., BARCONS C., HERNÁNDEZ H., ZALDÍVAR J.M. - Characterisation of an exothermic reaction using adiabatic and isothermal calorimetry - (accepted for publication in Journal of Thermal Analysis), 1992

[15] MORRIS S.D., BELL K., OSTER R. - Top-venting of flashing high-viscosity fluids - Chemical Engineering and Processing, vol. 31, pp. 297-305, 1992

[16] BELL K., MORRIS S.D. - Flow visualizations during top-venting from a small vessel - J. Loss Prev. Process Ind., vol. 5, no. 3, pp. 160-164, 1992

[17] BELL K., MORRIS S.D., OSTER R. - Vent-line void fractions and mass flowrates during top-venting of high-viscosity fluids - J. Loss Prev. Process Ind., vol. 6, no. 1, pp. 31-35, 1993

[18] MORRIS S.D. - Two-phase venting: New formulae for the average vapour drift velocity - US DIERS Users Group Meeting, Princeton, USA, 30 Sept.-2 Oct. 1992

[19] MORRIS S.D. - Top-venting of flashing high-viscosity fluids: vent-pipe inlet conditions (experiment and theory) - US DIERS Users Group Meeting, Princeton, USA, 30 Sept.-2 Oct. 1992

[20] MORRIS S.D. - Some comments on relief system inlet conditions during two-phase venting - 4th Meeting of ISO Technical Committee 185 (flashing flow through safety devices), JRC, Ispra, Italy, 7-8 April 1992

[21] MORRIS S.D. - Round-Robin calculation exercise-steam venting problem. 4th Meeting of ISO Technical Committee 185 (flashing flow through safety devices), JRC, Ispra, Italy, 7-8 April 1992

[22] KRÜGER G.J., HAUPT J., WEISS R. - A nuclear magnetic resonance method for the investigation of two-phase flow - In: Measuring techniques in gas-liquid two-phase flows, IUTAM Symposium Nancy, France, 1983, pp. 435-454, DELHAYE J.M., COGNET G., Eds. Springer, Berlin Heidelberg, 1984

[23] KRÜGER G.J. - Meßmethode zur Bestimmung einzelner Gemischkomponentenströme in Mehrstoff-Mehrphasen-Fluiden mittels kernmagnetischer Resonanz - Patentregistered at Luxembourg 24.1.1991, number 87879, prolonged 24.1.1992

[24] DUFFIELD J.S., FRIZ G., NIJSING R. - Critical Flow of a Chemically Reacting Two-Phase Multicomponent Mixtures - submitted to the International Journal of Multiphase Flow

[25] DUFFIELD J.S., FRIZ G., NIJSING R. - The Venting of Peroxide Solutions - 29th European Two-Phase Flow Group Meeting, Stockholm, June 1992

[26] STÄDTKE H., HOLTBECKER R. - A Hyperbolic Model for Inhomogeneous Two-Phase Flow - 29th Meeting of the European Two-Phase Flow Group, Stockholm, June 1 - 3, 1992

[27] STÄDTKE H. - On the Hyperbolic Nature of Two-Phase Flow Equations - EUR Report (in preparation)

[28] HOLTBECKER R. - Untersuchung Physikalischer und Numerischer Modelle für Inhomogene Zweiphasenströmungen - Universität Stuttgart, Ph.D. thesis (in preparation)

[29] STÄDTKE H., HOLTBECKER R. - Numerical Simulation of Two-Phase Flow Based on Hyperbolic Flow Equations - paper submitted to the 6th Conference on Nuclear Reactor Thermal-Hydraulics, NURETH-6, Grenoble, October 5-8, 1993

[30] ANDRONOPOULOS S., BARTZIS J.G., WÜRTZ J., ASIMAKOPOULOS D. - Modelling the effects of obstacles on the dispersion of denser-than-air gases - In preparation for the Journal of Hazardous Materials

[31] ANDRONOPOULOS S., BARTZIS J.G., WÜRTZ J., ASIMAKOPOULOS D. - Simulation of the Thorney Island Dense Gas Trial No. 8, Using the code ADREA-HF - Process Safety Progress, Vol. 12, No. 1, January 1993

[32] CARISSIMO B., ANDRONOPOULOS S., BARTZIS J.G., WÜRTZ J. - Intercomparison on Heavy Gas Dispersion Between the Three Dimensional Models MERCURE and ADREA. Part I: Instantaneous Release Without Obstacles - EDF Report RE-33/92-15

[33] STATHARAS J.C., BARTZIS J.G., WÜRTZ J. - Prediction of Ammonia Releases Using the ADREA-HF Code - AIChE Summer Meeting, San Diego, August 1992, accepted for publication in Process Safety Progress

[34] WÜRTZ J. - A One-Dimensional Shallow Layer Model for Dense Vapour Cloud Dispersion - Technical Note in preparation

[35] RUEL F. - Numerical Simulation of Reactive Gas Flows: Assessment Calculations of Transonic Multidimensional Flows - CEC/JRC, Ispra, Techn. Note No.I.92.01, January 1992

[36] RUEL F., HULD T., SORIA A. - Application of High Resolution Schemes to the Numerical Simulation of Reacting Gas Flows - Quatrième Seminaire sur les ecoulements de fluides compressibles, CEN Saclay, France, Janvier 1992

[37] HULD T. - Numerical Simulation of Reactive Gas Flows: Assessment Calculations of Deflagration/Detonations - CEC/JRC, Ispra, Techn. Note NO.I.92.66, June 1992

[38] RUEL F. - Partial equilibrium Solvers for Reactive Transonic Flows; Basic Algorithms - CEC/JRC, Ispra, Techn. Note No.I.92.125, November 1992

[39] HULD T. - Numerical Simulation of Reactive Gas Flows: Explicit and implicit solvers for diffusive processess - CEC/JRC, Ispra, Techn. Note (in preparation)

[40] SORIA A., PEGON P. - Semi-implicit conservative upwind schemes for transient compressible flows - EUR 14986 EN

[41] RUEL F., SORIA A. - A Semi-Implicit Solution of the Multicomponent Euler Equations in Subsonic Regime - CEC/JRC, Ispra, Techn. Note (in preparation)

[42] PETER G. - Adaptive Grids in Reactive Gas Flows - CEC/JRC, Ispra, Technical Note (in preparation)

[43] PETER G., DIGLIB A. - Device Independent FORTRAN Library for Graphic Primitives - CEC/JRC, Ispra, Technical Note (in preparation)

[44] BEST C., ROEBBELEN D. - TURCOM, Visualization of Reacting Gas Flows on Non-Structured grids - CEC/JRC, Ispra, Technical Note No.I.92.136, December 1992

SPECIFIC PROGRAMMES

1.6 REFERENCE METHODS FOR THE EVALUATION OF STRUCTURAL RELIABILITY

The activities in the field of structural safety are largely centred around the use of the new reaction-wall facility, now named ELSA, which was officially inaugurated on October 16 in the presence of Authorities and technical delegations from the Member States of the Community. The facility will initially be used for prenormative research in support of Eurocode n°8 (EC8), the European design code for structures in seismic areas.

A first integrated research programme, which concerns the earthquake behaviour of civil engineering structures, has been set up in close collaboration with the Directorate General III of the European Commission, the EC8 expert group, and a number of research organisations of the Member States grouped in the European Association of Structural Mechanics Laboratories.

The execution of this integrated programme will be performed jointly by the Association and the ELSA team now grouped within a scientific network under the Human Capital and Mobility (HCM) programme. Access to ELSA will also be facilitated through its recognition as a "Large Installation" under the HCM programme.

1.6.1 IMPLEMENTATION AND EXPERIMENTAL VALIDATION OF THE PSEUDO-DYNAMIC (PSD) TEST METHOD

During this year the three main tasks of the experimental sector in charge of the ELSA reaction-wall laboratory have been:

- Start-up of the laboratory.
- Conclusion of the research on the strain-rate sensitivity of reinforced concrete members.
- Qualification of the PSD method on a multi-degree-of-freedom steel frame.

In addition, work has been performed to further develop the data file management system for ELSA and to assess, through preparatory numerical studies, the applicability of the PSD method to structures with distributed mass.

Finally, the design and commissioning of a 4-storey reinforced concrete frame to be tested in ELSA in 1993 has been completed in collaboration with the European Association of Structural Mechanics Laboratories.

Start-up of ELSA laboratory

Once the civil engineering works of the ELSA laboratory itself were completed, work began on the running-in of the hydraulic and measuring equipment essential to the functioning of the PSD method.

As regards the hydraulic apparatus, shake-down tests have been conducted on the oil distribution system (up to 1.5 times the design pressure). Following this, a long duration flushing was conducted to remove impurities present in the oil circuit (piping, manifolds and pistons), in order to achieve the oil purity required for the safe operation of the servo-valves controlling the pistons. Twelve double-acting servo-controlled actuators have been received and extensively bench tested to calibrate the displacement and force measurement accuracy. The performance was found to be within the specifications.

A considerable amount of heavy auxiliary structural rigs and fixtures have been designed and constructed. These fixtures help to position the specimens into place. For the complete re-location of entire buidings, a lifting and transportation device has been designed.

Last year's annual report already presented some results on PSD experiments conducted on a 1.5 m reinforced concrete column treated as a single-degree-of freedom (DoF) system.

This year, the performance of the digital control system developed in house was verified on a 9m long beam, having a cross-section of 1.25x0.5 m, treated as a four DoF system. This type of experiment provides a difficult testing ground for the PSD controlling technique. This is because strong interactions between the pistons arise from the stiff coupling provided by the beam. This test setup will be further used for the comparison of different control and time integration algorithms.

Strain-rate sensitivity of reinforced concrete

From the results of an alternating frequency testing technique developed in house, it was possible to quantify the strain rate sensitivity of R/C structural members (beams and columns) from 0.002 to 0.2 Hz. This method was already presented in last year's report but, then, could not be used at frequencies higher than 0.2 Hz. The reason was that, given the size of the specimen and the level of deformation, the required hydraulic flow was not available. The testing apparatus was subsequently modified by introducing two accumulators and two servo-valves working in parallel to provide the peak oil flow rate required at 1Hz.

It was found that by increasing the loading rate from 0.2 to 1 Hz did not increase significantly the measured load on the specimen. In conclusion, we have found that in the range 0.002 to 1Hz the measured load increases by about 5-6%. Given that this range of testing frequencies is typical of that between PSD and real-time seismic inputs, it is thought that the PSD tests to be conducted on full-scale buildings will not be significantly affected by the reduced speed of deformation inherent in such a testing technique.

The results of the investigations have been accepted for publication and the proposed test procedure is currently being tried out by other laboratories [1].

Tests on a three-storey steel frame

The objectives of this research are on the one hand to act as a final check of the implementation of the pseudodynamic test method on the Reaction Wall and on the other hand to study the seismic behaviour of a realistic steel frame structure designed according to the relevant Eurocodes for steel constructions.

A full scale three-storey one-bay steel frame has been constructed in the ELSA Reaction Wall Laboratory [2], (*Fig. 6.1*). The objectives of this research are on the one hand to check the implementation of the pseudo-dynamic (PSD) test method [3] and on the other hand to study the seismic behaviour of a realistic steel frame structure with semi-rigid joints, designed according to the relevant Eurocodes for steel construction. The resulting experimental data will be compared with the analytical predictions from a previously developed computer model [4, 5, 6].

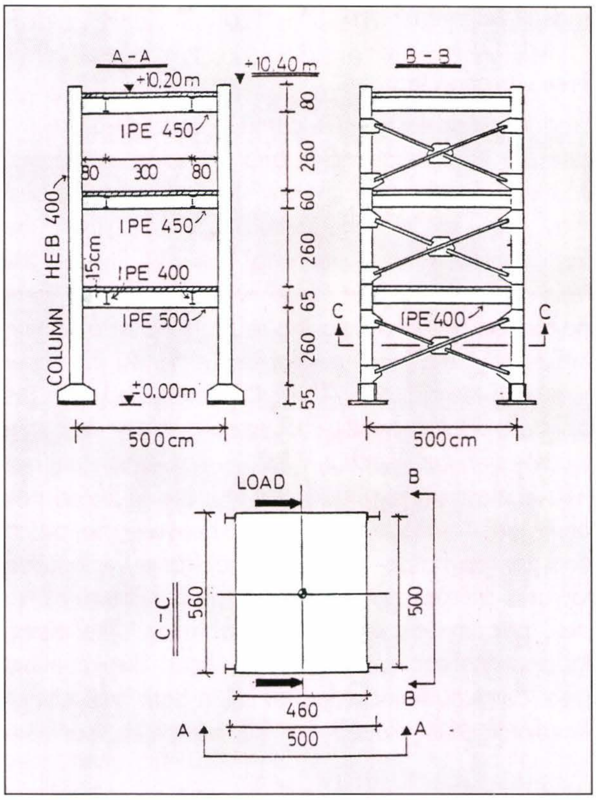

Fig. 6.1 Three-storey steel frame

Test - structure

The steel frame was designed using the relevant Eurocodes 3 and 8 [7, 8] for steel structures (*Fig. 6.1*). Actual materials were used for the construction and all components of the steel structure were made in a steel fabrication shop. It was then assembled on site. The steel frame was subjected to simulated earthquake induced loads using the PSD test method.

The basic criterion for the design and construction of the beam to column joints and connections was to select simple configurations, typically used in earthquake moment resisting steel frames. Hence, a welded beam to column joint and connection was selected.

Structural stiffness evaluation

The elastic stiffness properties of the structural system were determined by implementing the quasi-static testing mode of the control system. The direct stiffness approach was utilized by imposing a displacement cycle at one DoF at a time, while keeping the displacements at all other DoFs equal to zero. The resulting restoring forces developed by the specimen were measured by the load cells associated with each DoF. ***Fig. 6.5*** shows the displacement pattern of floor 3 while the displacements of floors 1and 2 are maintained equal to zero. In this case the frame top was displaced from 0 to 2.5 mm, unloaded and then the excursion was repeated for negative displacement of equal magnitude. This complete cycle was repeated, then the displacement amplitude was doubled and the cycle was repeated again. Note that the displacement wave has been smoothed to avoid perturbations caused by discontinuities.

In the elastic range the model was indeed assumed to behave linearly, and the stiffness terms were determined by means of least-squares approximations of the measured force-displacement relationships. So the slopes of the force-displacement diagrams give the stiffness terms Kij where subscript i indicates the measured load cell while j gives the actuator used to impose the displacement on the specimen. The stiffness matrix was found to be slightly assymmetric.

Linear elastic earthquake response

The main purpose of this test was to verify the proper implementation of the PSD test method, in particular the overall behaviour of the integration algorithm, distributed digital control system, instrumentation and data acquisition system before addressing more complicated non-linear problems.

A second set of PSD-tests on the steel frame were executed using a real earthquake signal at a load level insufficient to introduce yielding in the structure. The signal from the Kalamata Greece- (southern Peloponnessos) earthquake (***Fig. 6.6***) from September 1986, having a duration of 10 sec has been used. The steel frame had been designed to resist an earthquake of that type. The earthquake signal was scaled down by 50%.

The steel frame structure under testing represents a part of a real building. The complete structure, taking into account the real masses and the steel skeleton stiffness would have a period of 0.75 sec. In order to simulate a realistic situation, the steel frame's fundamental period of vibration has been "set" to be 0.5 sec, thus taking into account the stiffness of partition walls and non-structural elements which in reality increase the stiffness of a building. It is a fundamental advantage of the PSD-method that any mass in a structure under testing can be simulated numerically. Hence, this option has been fully exploited.

The displacement and force response history of the structure excited by the low level earthquake is shown in ***Figs. 6.7*** and ***6.8***. From the measured displacement versus force relationship it was observed that the test specimen behaved approximately elastically.

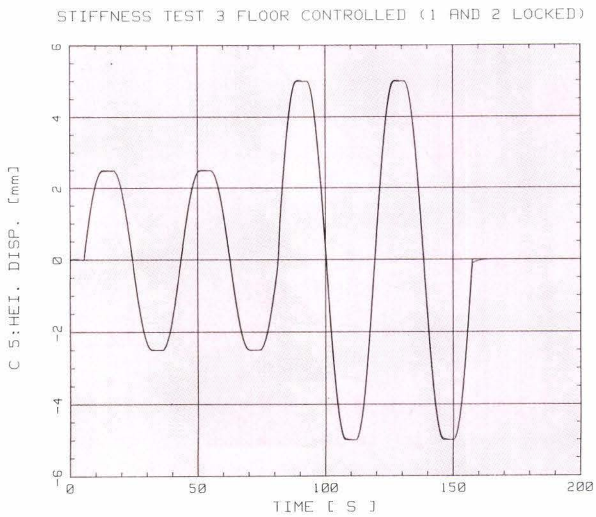

Fig. 6.5 Imposed displacement history for stiffness matrix evaluation

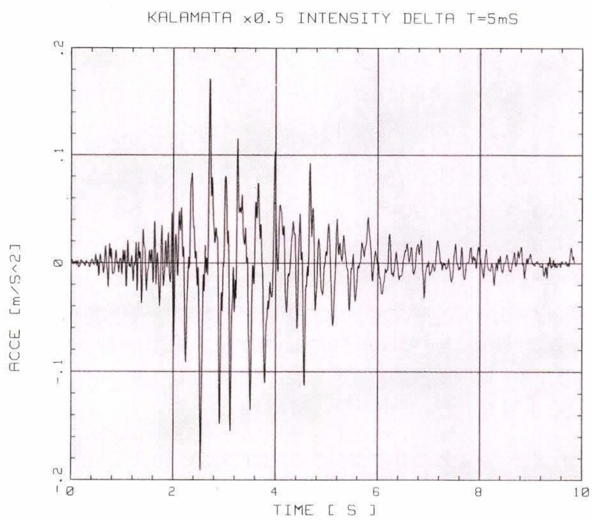

Fig. 6.6 The half-intensity 1989 Kalamata earthquake used as input acceleration signal

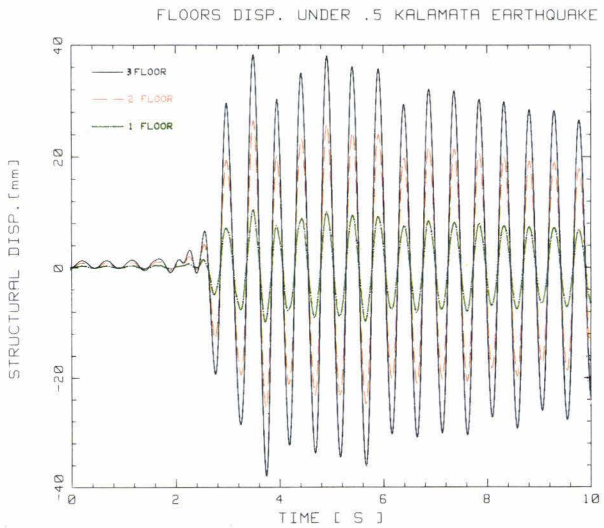

Fig. 6.7 Structural displacements, at each of the three storeys of steel structure, resulting from the sample earthquake of Fig. 6.6

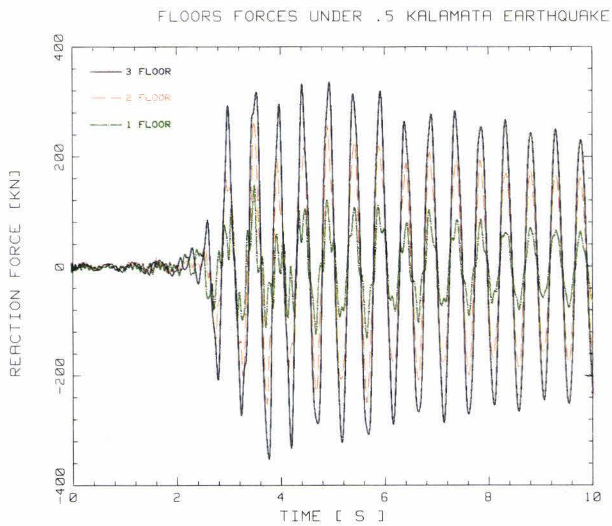

Fig. 6.8 Piston forces at each storey resulting from the sample earthquake of Fig. 6.6

A preliminary analysis of the obtained results has shown that the steel frame girders possess a high in-plane bending stiffness due to the presence of a reinforced concrete slab of 15 cm thickness. However, the welded beam-to-column joints and connections contribute to the horizontal frame deformation, thus increasing the inter-storey drift. This result shows the need of capturing this effect in the European building code for a seismic construction - (Eurocode 8). A numerical simulation with a computer model already developed [4], using the DRAIN-2DX computer programme [10] was found to be in good agreement with the experimental behaviour.

Subsequently, the earthquake intensity will be increased to introduce severe plastic deformation of the structure. During this testing phase the seismic energy absorption mechanisms and the ductility capacity of the steel frame will be studied.

Experimental data file management system

The ELSA laboratory will produce a very large number of experimental files. It is therefore important to manage them in the best way and make their content accessible both to the computational engineering group and to other laboratories.

The experimental files are based on a specific standard called the ELSA Experimental Data Files (EEDF) for internal use only. The communication through the Safety Technology Institute and the Computational Engineering Group will be based on the Experimental Data File (EDF) standard common to the whole Institute. The managment of the EEDF files and the interface with the EDF standard data bases is realized by means of the programme Working on Data Base (WDB) developed by the ELSA team.

Each test, performed in the ELSA laboratory, will produce many EEDF files, each one being related to one signal. For this reason WDB has been conceived for an easy handling of a large number of files. The WDB programme has been developed under the Unix environment in Fortran with some specialized routines written in the C language; formally it works as a Unix command accepting metacharacters and standard I/O operations.

The basic functions of the WDB programme, shown in *Fig. 6.9*, are the following:

- Inquiry of the EDF data bases.
- Interface between the EDF data bases and the EEDF data base with the possibility to transfer files from one to the other of the two standards maintaining full compatibility during the transfer (no information will be lost).
- Inquiry of the EEDF data base.
- Possibility to convert binary EEDF to ASCII files (and vice-versa) which can be considered as formatted images of the EEDF standard.
- Possibility of alphanumerical and arithmetical operations on the EEDF files in order to produce files with optimized information to be stored in the EDF data bases. These operations are related to

the definition of the variable names and units, to the change of unit of measurement, time step, title and other characteristics.
- Possibility to read non standard ASCII files in standard EEDF files (and viceversa).

This last function has a high flexibility in reading or writing ASCII files whatever their specific format is. This function has been implemented to ease the possibility to transfer data between the ELSA laboratory and external ones. The only condition is that the files used for the data transfer must be written in ASCII format.

Extension of the pseudodynamic method to test distributed mass structures

In the pseudodynamic (PSD) method, at each time step the dynamic deformation of the structure, computed by solving the equation of the motion for a given input signal, is reproduced in the laboratory by means of actuators attached to the sample at specific points. The reaction forces at those points are measured and used to compute the deformation for the next time step. As the reaction forces are known, and as the knowledge of the stiffness of the structure is not needed, the method can be effective also for deformations leading to strong nonlinear behaviour of the structure [11]. On the contrary, the mass matrix and the applied forces must be well defined and known. For this reason the PSD method can be applied without approximation when the masses can be considered as lumped at the testing points of the test structure.

The possibility to extend the PSD method to test structures with distributed mass is very important from an engineering stand point and is currently under investigation at the JRC-ELSA laboratory [12]. During a pseudodynamic test on a lumped mass structure (*Fig. 6.10*), controlling all degree of freedom, at time t the displacements x are imposed and the reaction forces r are then measured. These are used in the equation of motion at the next time step.

Supposing, for example, to execute a PSD test controlling only the top point (P4), the masses not controlled by the actuators can be considered as distributed. In this case (*Fig. 6.11*), the static deformation imposed in laboratory is not the same as the real dynamic deformation. In particular, executing a PSD test in this way, the reaction force is known only at P4, while in the mathematical model the reaction forces are distributed on the four points. One way to solve this problem is to build an equivalent model of the structure under examination condensing the masses to the points controlled during the test.

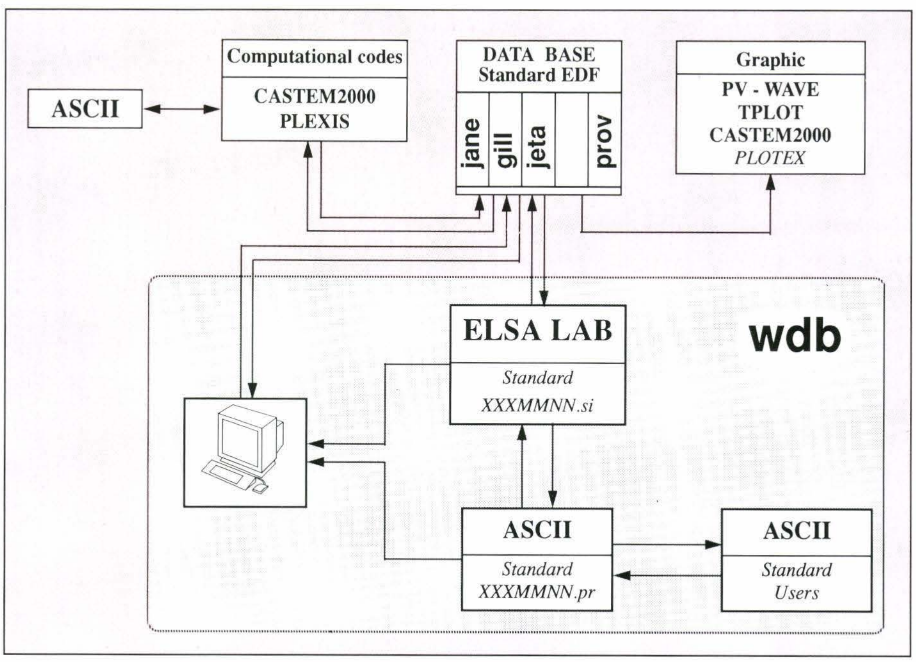

Fig. 6.9 EEDF management system

A mathematical model has been set up to reduce the degrees of freedom of a finite element (FE) representation of a real structure to the few degrees of freedom, which are the experimental displacements, through the use of a rectangular matrix that will be called the condensation matrix. Using this matrix the general dynamic equation of motion can be condensed to give a "Reduced Degree of Freedom" (RDF) equation with an equivalent PSD formulation. The condensation matrix is computed using normalized static deformations of the structure obtained by the application of unitary forces in the points of interest.

A numerical procedure, called "pseudodynamic simulation", has been set up to reproduce a PSD test using a condensed model of the structure. In this way it is possible to verify the accuracy of the RDF model, comparing the results of a FE dynamic analysis of the structure considering all degrees of freedom and the results of the PSD simulation based on the RDF model of the structure.

The PSD simulation consists of the following steps:

- Solution of the equations of motion using the general FE model and acquisition of the displacements in the experimental points;
- Computation of the statical deformation of the structure imposing the congruence with the displacements;
- Computation of the reaction forces in the experimental points;
- Solution of the PSD condensed equations assuming for the reaction forces those previously calculated.

The numerical simulation of a planned PSD test allows the execution of parametric studies in order to define the better choice for the position of the experimental points. The method has been applied to a numerical simulation of the behaviour of a realistic and complex structure [13] with distributed mass consisting of a masonry building with two floors (***Fig. 6.12***). The FE model consists of about two thousand degrees of freedom and the condensation has been made for four testing points (V1 to V4 in ***Fig. 6.13***).

A linear elastic dynamic analysis using a realistic seismic input (El-Centro Earthquake) has been performed with the general FE model and the reactions of the structure have been recorded in a file and used as input for the PSD simulation with the four degrees of freedom model (the reaction force at point V2 is shown in ***Fig. 6.14***). The displacement vs. time of the point V2 from the general FE model and from the PSD simulation is plotted in (***Fig. 6.15***).

The comparison between the results showed a very good agreement suggesting that, at least for linear structural behaviour, the standard procedure proposed can evaluate the equivalent mass matrix and external force vector to extend the PSD method to test structures with distributed mass and shows the applicability of the condensed model. Work is in progress to extend the method to nonlinear structural behaviour.

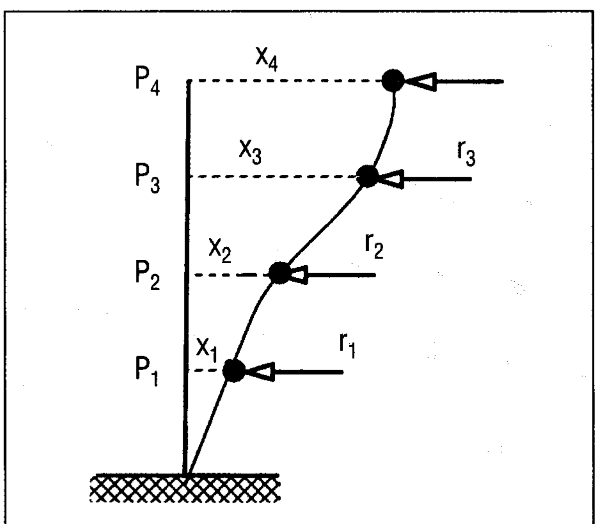

Fig. 6.10 Test on a lumped mass structure

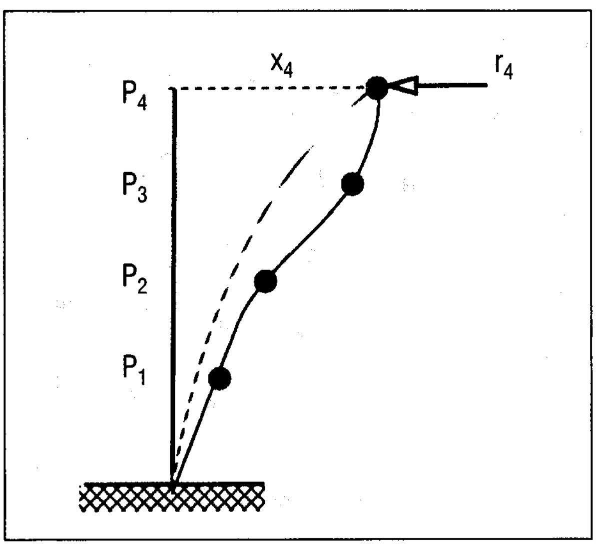

Fig. 6.11 Test on distributed mass structure

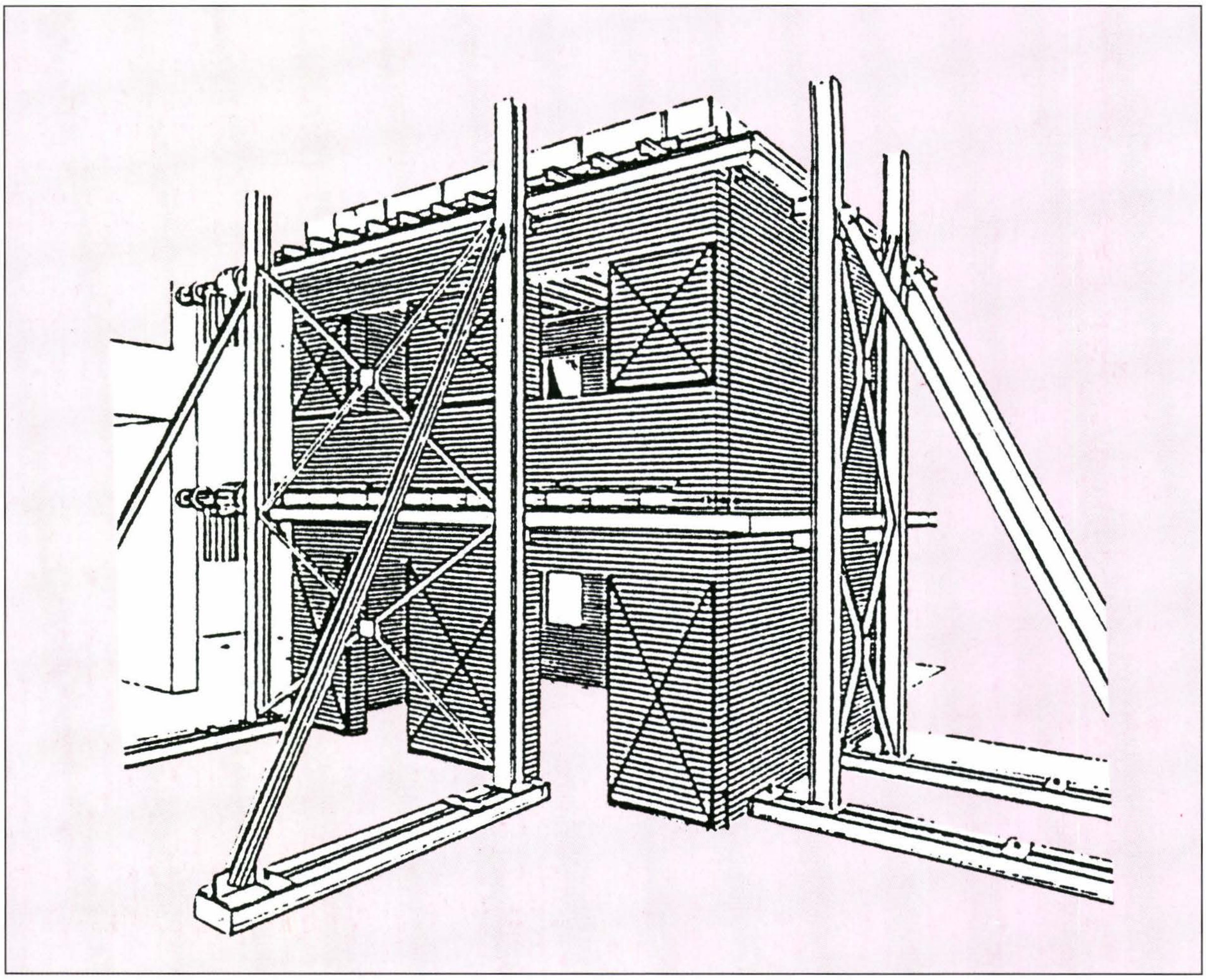

Fig. 6.12 Two floors masonry building

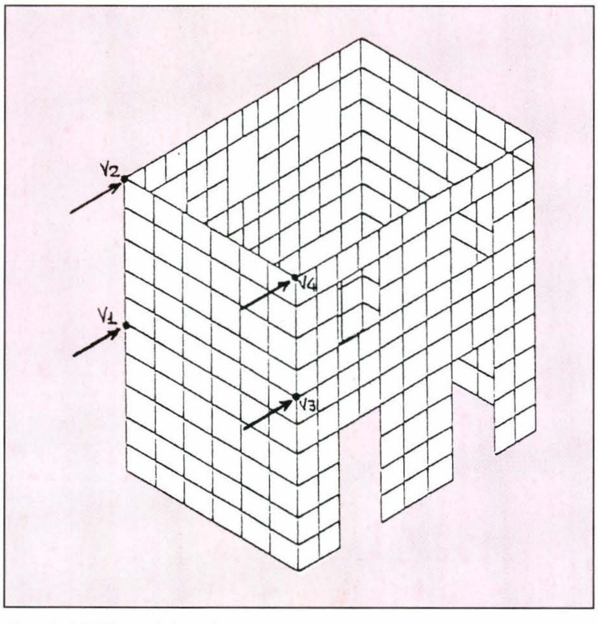

Fig. 6.13 FE model and testing points

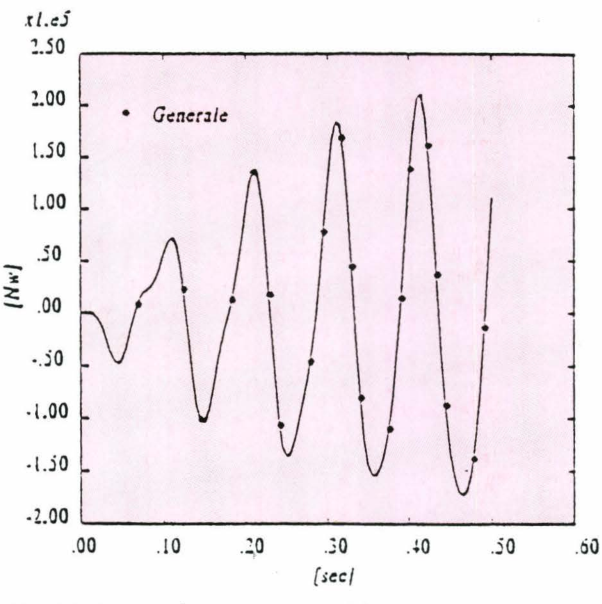

Fig. 6.14 Reaction force in V2 (FE model)

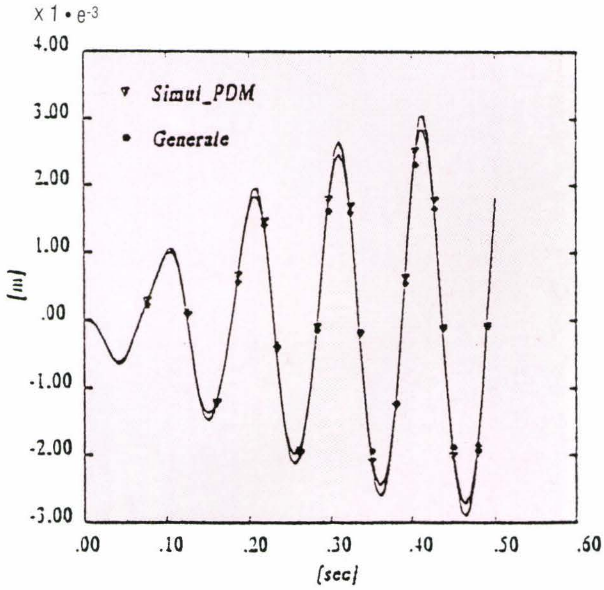

Fig. 6.15 Displacements in VS (FE and PSD)

Design of a four-storey reinforced concrete frame

A cooperative programme on the seismic response of reinforced concrete structures is currently being carried out in cooperation between the members of the European Association of Structural Mechanics Laboratories (EASML).

This research represents the continuation of a previous activity oriented to the seismic design of reinforced concrete structures according to Eurocode 8. In the first phase, the scope was to check the applicability of the code to real structures, to enable a comparison of the computer codes used in seismic analysis in different countries and to identify the research needs and priorities to make progress in the solution of specific problems still open in seismic design.

The purpose of the current phase is the definition of damage indicators and failure criteria and it is based on a testing activity which will result in a full scale test to be performed in the ELSA facility. The definition of the building to be tested as well as the preliminary design has been carried out by the members of the reinforced concrete working group of EASML, whilst the final design and all the construction procedures are in the hands of the ELSA team.

The building is a four-storey framed structure (*Fig. 6.16*), with plan dimensions of approximately 10x10 m and a total height of 12.5 m. The plan is symmetrical with respect to the direction of testing, but eccentric in the orthogonal direction for possible subsequent tests after repair. The specimen will be represented by the bare structure, however, provisions have been made for adding infill panels for subsequent tests. The building has been designed for typical live loads and for severe seismic actions (ground acceleration = 0.3 g, medium soil conditions). The design actions have been defined as for "Ductility Class High" in the Eurocode, so significant inelastic deformation is expected during the test.

The materials have been defined as ordinary C25/30 concrete, whilst for the reinforcing bars the Tempcore B500 steel has been selected. This steel, originally not included in Eurocode 8 provisions, is becoming dominant in some European countries, so the importance of assessing the adequacy of this material for earthquake resistant constructions has been recognized.

The detailed design exercise has resulted in the drawings necessary to tender for the construction which will begin in early 1993. The final design included all the necessary design solutions and checks for the transportation of the specimen and the introduction of loads. The specimen will in fact be

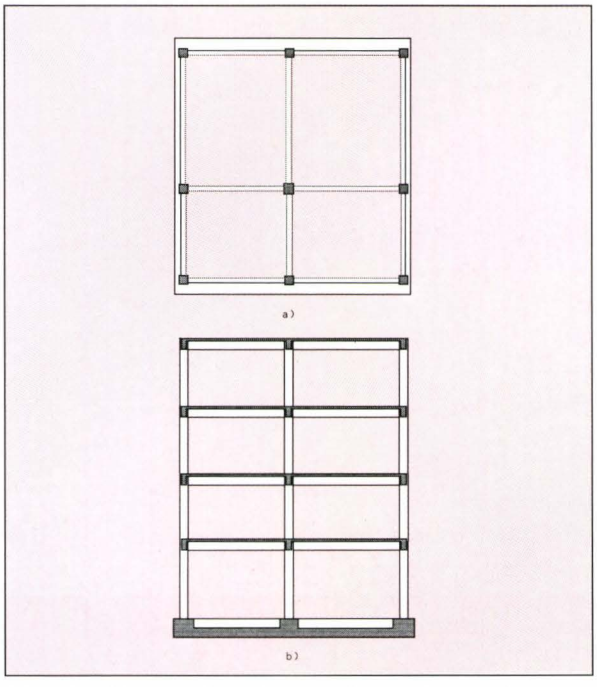

Fig. 6.16 4-Storey reinforced concrete frame to be tested in ELSA: a) plan, b) elevation

constructed in the area outside the ELSA laboratory. After the completion of the construction, the specimen will then be raised from the floor by hydraulic jacks and moved inside the laboratory by means of plastic tube rollers.

These transportation procedures had to be studied in great detail due to the huge weight of the specimen (more than 500 tons). However, this will reduce the construction problems and allow other tests during the construction phase. The need for this transportation procedure has determined the stiffness and strength of the foundation slab. Special provisions were also necessary for the introduction of the loads. Two actuators will be attached to each floor in the direction of testing, whilst one additional actuator per storey will control the orthogonal displacement.

The test on this four-storey building will represent the first major pseudodynamic test on a realistic, full-scale, reinforced concrete structure designed according to the Eurocodes. Great interest in the results has been expressed by the scientific community, in particular by the experts in charge of the calibration of the Eurocodes.

Definition of damage indicators and failure criteria

The main objective of the test on the full-scale four-storey reinforced concrete frame building is to contribute to the definition of the damage indicators to be used for defining the response of the structures to seismic actions.

A safety assessment for earthquake loads is in fact much more difficult than for monotonic loads, because the resulting seismic forces depend on the stiffness and strength of the structure, and because the performance of the structure cannot be defined by simple parameters. The definition of a convenient damage indicator is therefore necessary both for safety checks of particularly important structures and for the calibration of the codes for construction in earthquake prone areas.

The JRC, already active in the research on this topic [14] has participated in the task of defining the most appropriate parameters to be assumed as damage indicators. A first contribution [15] is represented by preliminary non-linear analyses of the behaviour of the building. In this study, the behaviour of the building subjected to various artificial accelerograms (which approximate the response spectrum given in the code) has been analysed, both using standard computer codes for the damage analysis of buildings and models developed in CASTEM 2000. The analysis has been carried out for different intensity levels, to assess the adequacy of the various damage indicators to represent the local and global damage of the structure.

Another study [16] has dealt with the definition of a damage indicator for describing the global performance of the structure. In this respect, a damage indicator has been defined as a parameter which relates the damaged state corresponding to a dynamic history to the monotonic load deformation diagram. The adequacy of various proposed damage parameters was discussed, in particular the possibility to interpret the usual definition of ductility when a damage indicator has been studied. A definition of "equivalent ductility" capable of including, the maximum deformation, as well as the contribution to the total damage due to cyclic energy dissipation, were introduced. Response spectra in terms of equivalent ductility have been produced for artificial accelerograms (***Fig. 6.17***) as well as for a wide selection of real records. The equivalent ductility appears to be a convenient and simple parameter for defining the damaged state. However, in most cases of earthquake loading the contribution to damage due to the cyclic behaviour appears quite limited, so that the usual definition of ductility may be regarded as a sufficiently accurate damage parameter.

Dynamic characterization of test structures

A preliminary, but essential, phase of the testing of any structure is its dynamic characterization. This includes the measurement of its masses, stiffness characteristics, damping properties and natural frequencies.

The techniques to measure these parameters have been calibrated in measuring the properties of the reinforced concrete columns which are intended to be used for implementing alternative formulations of the pseudodynamic test method [17], and of the three-storey steel frame structure described previously.

The stiffness matrix has been measured both using the "flexibility" approach (i.e. by applying a displacement or a force to the selected degree of freedom, measuring the displacement in the other points) and with the "stiffness" approach (i.e. constraining all the degrees of freedom except one, and measuring the restoring forces). The stiffness approach turned out to be by far the most accurate.

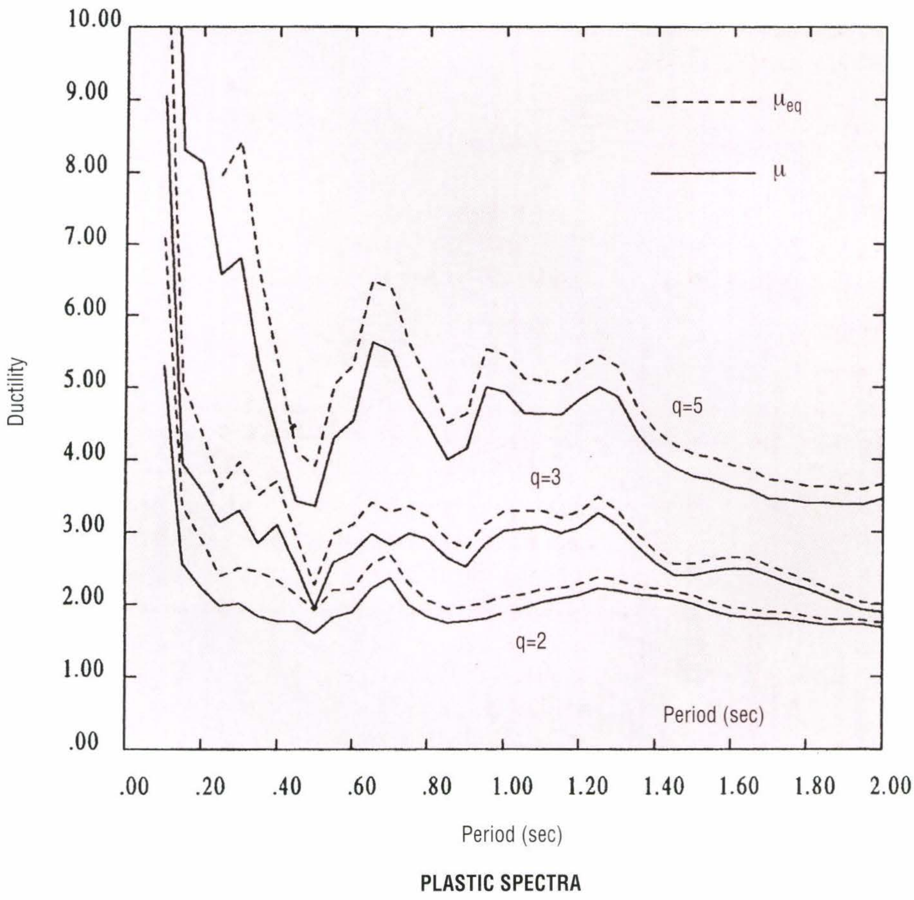

Fig. 6.17 Response spectra in terms of "equivalent ductility"

The masses, the damping properties and the natural frequencies have been measured by means of free vibration tests. The reinforced concrete column was set into vibration by hammering, whilst the steel frame was excited by a snap-back test, i.e. it was deformed by stressing up to rupture a steel bar attached between the specimen and the reaction wall. The dynamic properties were measured both with a signal analyzer and with the data acquisition system of the laboratory. Subsequently, pseudo-dynamic simulations of the free vibration test were performed, checking both the accuracy of the dynamic characterization and of the implementation of the pseudodynamic test method.

1.6.2 MODELLING AND STRUCTURAL ANALYSIS

The activity during 1992 was mainly directed to the further refinement, calibration and exploitation of analytical models for simulating the earthquake response of civil engineering structures. In addition, a strong effort has been made in order to promote the ELSA facility and its modelling activities internationally.

Taking advantage of the venue of the 10th World Conference on Earthquake Engineering (10WCEE) in Europe (Madrid, July 1992) a series of papers [18, 19] were presented there. In addition, participation in national and international research programmes in the field of Earthquake Engineering was persued. Special reference is made to the activities of the working groups of the EASML and the SEISMIER programmes. Moreover, new collaborations were set-up during the present year, namely: a) HCM - Network on Pre-Normative research in support of EC8, b) The COST-C1 action on the study of semi-rigid civil engineering connections, c) The participation in the Italian programme on the study of masonry buildings (GNDT: Task - Prevention of damage to buildings, Sub-task - Experimental evaluation of the seismic behaviour of structures). ELSA will perform analytical work and PSD tests on several unreinforced masonry panels.

Quantification of earthquake input for seismic analysis and testing

The main aims of the seismic tests on physical models are the assessment of design codes, qualification of innovative design concepts and construction technologies and the calibration/ assessment of analytical models.

The test specimen, in most cases, is designed according to a code or it is to be qualified following determined norms. For this, a design seismic action is prescribed for the structure. Thus, the earthquake signal to be used in the test must comply with the requirement that it should have a response spectrum as close as possible to the design response spectrum. Furthermore, the earthquake signals should reproduce the characteristics of real earthquakes, in particular their non-stationary characteristics both in time and in frequency.

Appropriate models to represent earthquake ground motion complying with the above requirements have already been developed and implemented in the computer code CASTEM 2000 [20]. These models have now been used to define the seismic input for the analysis of the 4-storey R/C structure to be tested in ELSA and will be used to perform the pseudo-dynamic tests. The earthquake signals, shown in ***Fig. 6.18***, were generated in accordance with two requirements, namely complying with the EC8 spectrum for Soil B and peak ground acceleration 0.3g, and having non-stationary characteristics identified in the Friuli earthquake.

Structural model development

Accurate calculation of the non-linear response of complex civil engineering structures to cyclic loading is currently not practicable. To improve this situation, modelling efforts are being made at three levels:

- local models describing the detailed material behaviour,
- member models encapsulating this behaviour for structural elements,
- combinations of such members to simulate complete structures.

Local modelling of masonry

In the continuation of the work performed in 1991 (derivation of the macroscopic linear elastic characteristics of masonry through the homogenization theory), the scope in 1992 was to investigate the nonlinear behaviour of masonry using the same technique (homogenization) within the following two complementary nonlinear frameworks:

- limit analysis
- damage modelling

Limit analysis : biaxial failure criterion of masonry

Within the limit analysis framework, the homogenization theory allows the derivation of the macroscopic plastic failure criterion of a composite material alone from the knowledge of the plastic failure criteria of the constituents. Being approximate in the case of masonry (plastic failure criteria are rather optimistic for materials such as brick and mortar), limit analysis was expected to give quick but only crude results for the macroscopic biaxial failure criterion.

The lower-bound method has been implemented in CASTEM 2000. From a numerical point of view, it amounts to maximizing a linear function (macroscopic stress) under linear equality constraints (equilibrium conditions) and nonlinear inequality constraints (failure criteria of the constituents). Since no operator in CASTEM 2000 could handle that specific optimization problem, it was decided to use a standard simplex method (operator SIMPLEX in CASTEM 2000), each non linear inequality being purposely approximated by a set of linear inequalities (in other words, the failure surface is approximated by a multifaceted surface).

The first tests were performed under the plane stress assumption in order to limit the size of the problem. Furthermore, a suitable choice of unknowns allowed a significant reduction in the number of variables and equality constraints. Nevertheless, the extent of the problem remained important even for a moderate number of mesh elements: for each triangle, there are 6 variables, 5 equalities and 3n inequalities if each failure criterion is approximated by n planes (typically, n is about 30 for a correct approximation of a Mohr-Coulomb criterion in plane stress). As a consequence, round-off errors accumulate and may become critical since equality constraints are

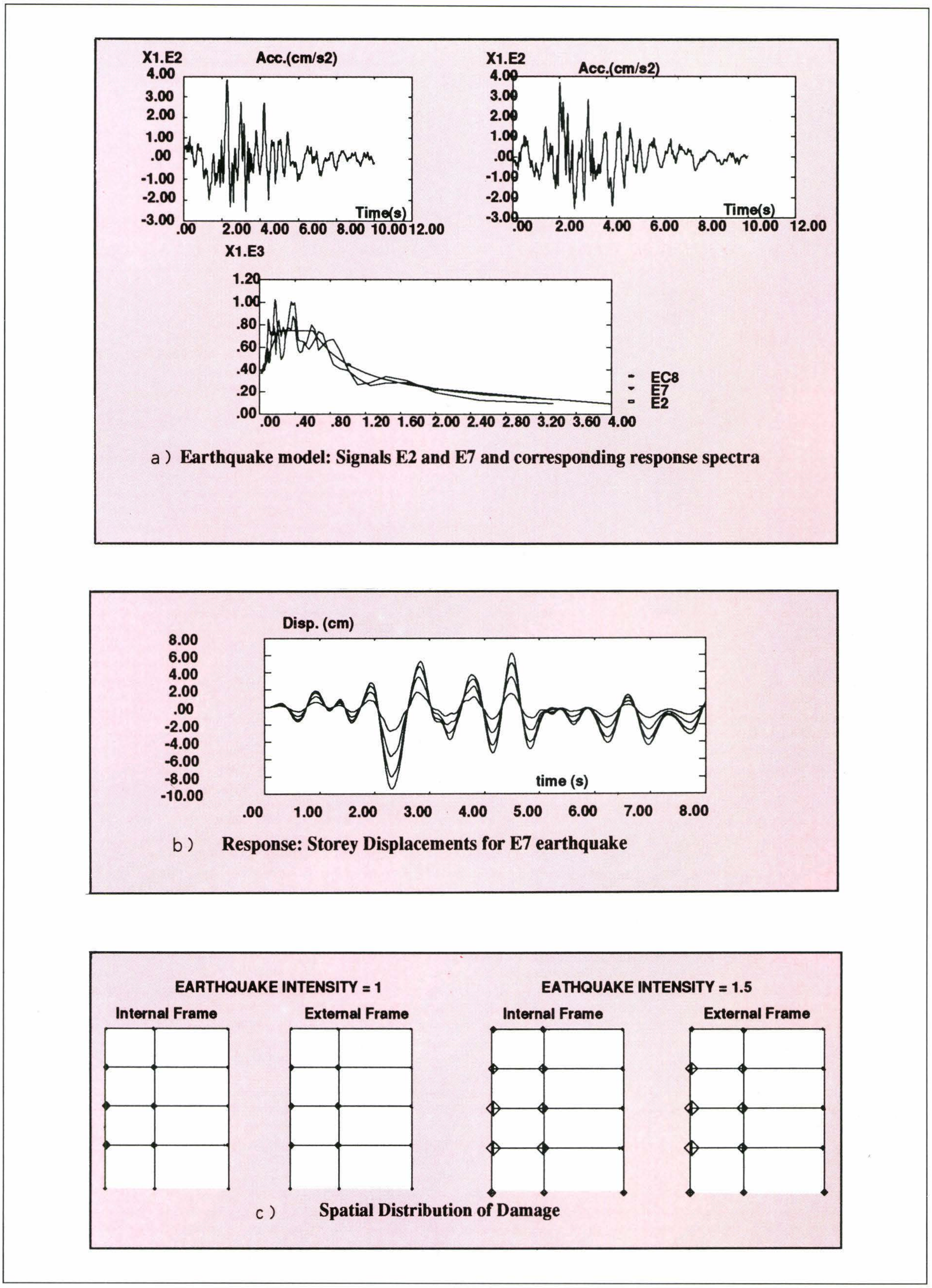

Fig. 6.18 Analysis of the R/C structure to be tested pseudo-dynamically in the ELSA laboratory

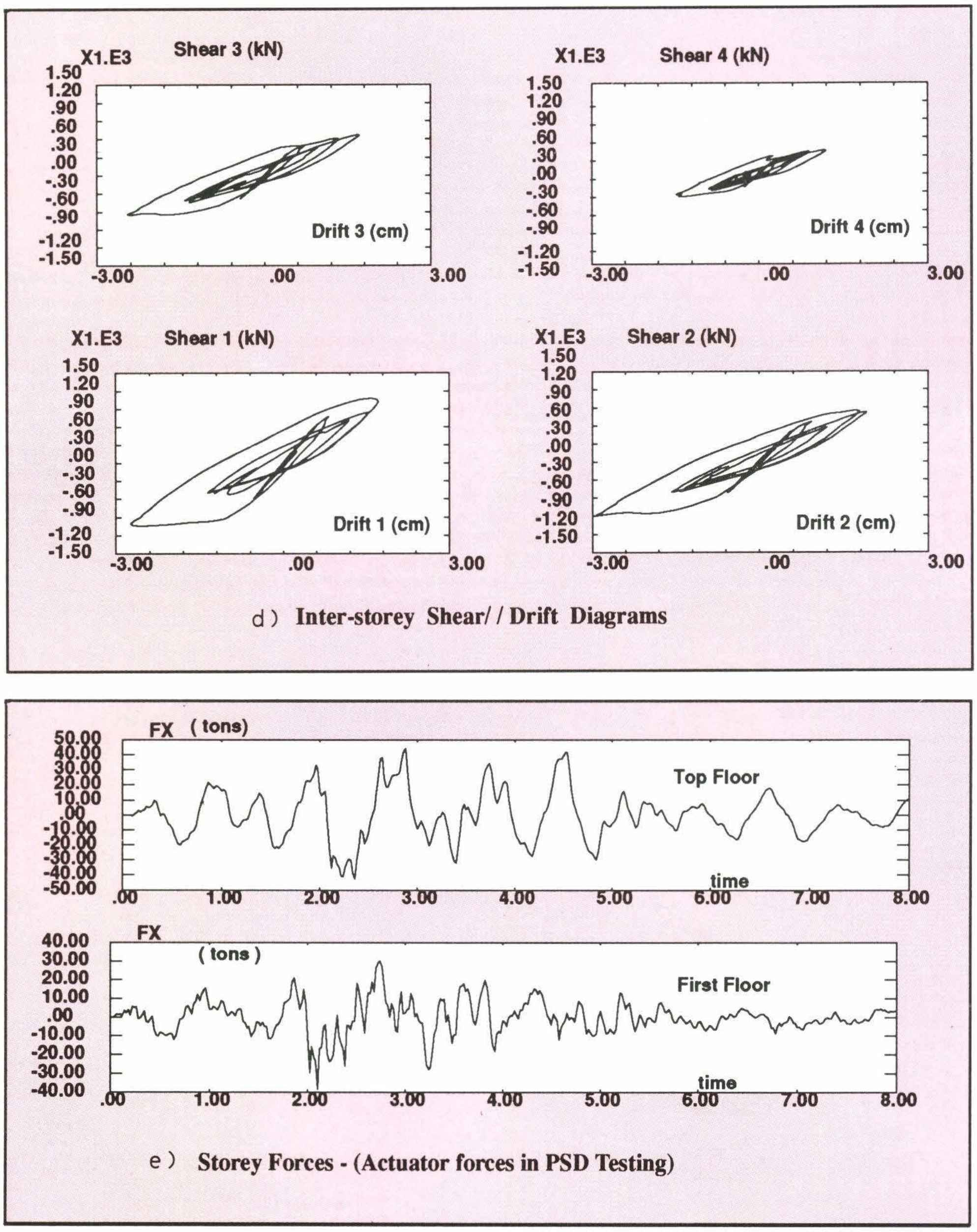

Fig. 6.18 Continued

generally redundant (because of numerical inaccuracy, redundant equalities are understood as incompatible equalities). A more specific optimization technique (able to handle large problems with redundant equality constraints and, possibly, nonlinear inequalities) is therefore required before proceeding to three dimensional problems.

Damage modelling: macroscopic behaviour law of masonry

If brick and mortar are both assumed to be subjected to damage, the homogenization theory leads to a complex macroscopic law because it is not possible to remove the information at the microscopic scale:

the macroscopic stress-strain relationship has an infinite number of internal variables (the whole microscopic strain field). In particular, the use of such a behaviour law would require a formidable numerical effort. At each Gauss point, a complete non-linear boundary-value problem must be solved on the basic cell (the smallest volume which generates by periodicity the entire structure of the composite). In any case, such a degree of accuracy being inappropriate in the case of masonry, it was found more suitable to elaborate a simplified macroscopic behaviour law able to reproduce the salient features of the behaviour of masonry.

In order to identify those features, computations have been carried out along relevant monotonic loading histories (increasing uniaxial vertical compression and increasing shear under constant uniaxial vertical compression).

The scalar damage model available in CASTEM 2000 was used. Both local and non local calculations were performed, the latter being mandatory in case of softening and localization to remove mesh dependency of the results.

Special attention was paid to the plane stress assumption: in the elastic range, 2D and 3D computations gave very close values for the macroscopic elastic constants, even though the plane stress assumption is scarcely valid for the 3D microscopic stress field. In the nonlinear range, however, it was found that 2D computations can lead to erroneous results, quantitatively (*Fig. 6.19*), as well as qualitatively (*Fig. 6.20*).

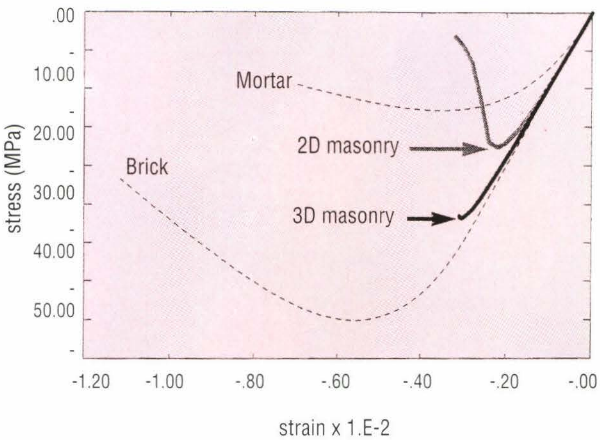

Fig. 6.19 Uniaxial vertical compression curves (vertical mortar joints were disregarded): 2D and 3D calculations coincide in the elastic range but diverge strongly in the non-linear range

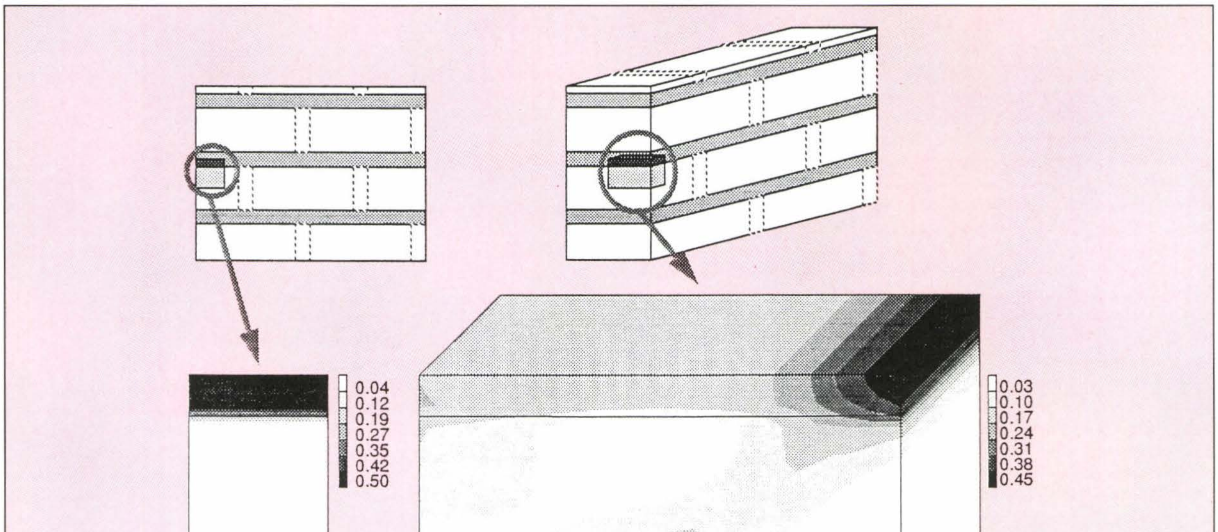

Fig. 6.20 Damage at maximum compression stress:
- 2D calculation (left): compression failure (crushing) occurs in mortar whereas the brick remains elastic
- 3D calculation (right): tensile failure (splitting) originates in the brick whereas the mortar is only partially damaged in compression (it was noted that, for the same damage index, strength degradation is greater and more rapid in tension than in compression)

Support to the implementation of new global or semi-global models

Development and use of global models are important for the reaction wall project since this level of modelling allows to reduce the number of degrees of freedom needed for describing a building and therefore allows to perform full-scale linear or nonlinear building calculations. For example, a global model may be used for describing an interstorey behaviour but also, depending on the degree of accuracy of the study, a beam/column joint behaviour.

In order to explain how the modelling is organized in CASTEM 2000, and how for instance a new model can be implemented, a document has been prepared [21]. In parallel, and in order to have only one integration point in the nonlinear beam elements which support the global models, a new beam element called TIMO (a Timoshenko element with linear approximation) has been implemented [22]. A preliminary study for the design and the implementation of a fiber model has also been realized.

Specific procedures/operators for the computation of the dynamics of nonlinear frame structures

The computation a of full frame building is performed using, for each column/beam a three-element member model, say an elastic beam with two small nonlinear beam elements supporting a global model at the end. The discrete dynamic equilibrium of such a system, with mass concentrated, as usual, at the end of each column/beam was solved using an implicit algorithm. The algorithm was chosen to be implicit in order to be able to equilibrate the internal nodes of each column/beam macro-element, which are mass free. In order to reduce the size of the nonlinear problem and to have an opportunity of introducing "explicit" time integration algorithms, it is foreseen to consider each column/beam as single (macro) element and to develop methods able to eliminate the internal nodes (nonlinear condensation).

A feasibility study has been started: some nonlinear condensation techniques had been developed at GIBIANE level (say at the level of the internal language of CASTEM 2000 which is applied as user interface) and applied in conjunction with two "explicit" time integration algorithms. The results obtained were very similar to those obtained with the implicit method, at least when the number of macro-nonlinear elements was low. However, rising this number led to a divergence process in the internal equilibrium of some macro-elements. Since the global law used was rather complex, employing two integration points for each nonlinear element, it was decided to simplify the element formulation (TIMO element [22]) and the global law (bilinear bending law proposed as illustration in [21]). New tests are currently performed using these two simplifications. In the absence of definitive results no attempt had been made to proceed to the further implementation of such condensation technique. As a side effect, the development of nonlinear condensation procedures offered the possibility of performing PSD tests on "numerical" nonlinear structures.

Automation of the numerical strategies for handling local damage models

The local modelling of concrete (damage mechanic approach) appears to be important for various identification studies of semi-global (or macroscopic) or global models, for example:

- Insight into the macroscopic behaviour of masonry (say a "semi-global" scale at which the various components are homogenized) using homogenization techniques along relevant displacement/stress controlled path histories.
- Study of cyclic behaviour of R/C beam/column joints and plastic hinge zones.

It was foreseen to identify in each case "typical" practical situations, and to use them as starting points for the development of (quasi) fully automatic path-following calculations procedures, limited in a first step to monotonic behaviour.

Some cases have been successfully treated in the case of masonry leading to very interesting results when comparing 2 and 3-D computations. Some developments had been performed for extending the possibility of non-local calculations (averaging of the local model over a volume in order to remove mesh dependency in case of softening and localization) to account for complex symmetry or periodicity.

The study of R/C beam/column joints and plastic hinges zones is only beginning. Although some "typical" practical situations have now been clearly identified at least in the masonry case, the effective use of path-following technique was not yet introduced.

Support to the use of optimization techniques

It was foreseen to use an optimization technique in order to performed a combination between limit analysis and homogenization for studying the limit behaviour of masonry. The use of a complex operator available in CASTEM 2000 was discarded in a first stage and a SIMPLEX (standard geometric iterative method for linear optimization) method was implemented. This method was tested on frame structures (say rods) and led to the required results. Applications to a continuum situation were also performed in 2-D configurations. It should be underlined that the number of variables and of redundant equalities (which introduces round-off disturbances into the SIMPLEX) increases very fast with the number of elements so that, at this stage, the solution method chosen appears to be rather ineffective.

Assessment of design code provisions

It is well known that improvements of seismic design codes and construction technologies should be based on a full exploitation of numerical models suitable to represent structural behaviour and give guidance for safety analysis, providing such models are calibrated and assessed on the basis of physical model tests.

The activities in ELSA are being developed along these lines and significant progress is being made. The analytical models developed and implemented in the reference computer code CASTEM 2000 [23] were used to perform the dynamic nonlinear analysis of the R/C structure to be tested at the Reaction Wall. Design and analysis of this structure were included in the research programme of the R/C Working Group (RCWG) of the European Association of Structural Mechanics Laboratories. *Fig. 6.18* illustrates the main steps of this study and more detailed information can be found in the contribution to the RCWG report [15]. In addition to the conventional results obtained from an analysis of this kind detailed information on the expected structural damage was obtained. Moreover the expected force history in the actuators was delivered. The knowledge of the frequency content of the forces is very useful in preparing PSD testing because it allows an appropriate set-up of the control system.

In addition, such models and others already available are being used to perform studies on structures designed in accordance with the Eurocodes. Giving continuity to the study of hybrid structural systems started last year [24] a new aspect has been covered in recent studies. Comparative analysis of hybrid and frame structures were performed in order to find design guidelines for different seismic regions and soil conditions. The study was conducted on the basis of the same structural reliability level, in terms of seismic performance, and in terms of economy. The main conclusions of that study included in a paper [18] presented in the 10WCEE were that ductile structures designed in accordance with EC8 have a good seismic performance and are economic. However, structures with high natural frequencies (greater then 2-3 Hz) should be designed as medium/low ductile systems in order to be safe and economic. In addition, it was verified that hybrid structural systems present some important advantages namely the strategic relocation of damage zones and prevention of collapse due to torsional effects. *Fig. 6.21* presents the average "economy factor" (linear design cost over ductile design cost) for different soil conditions and two earthquake ground motions.

Studies on the seismic behaviour of irregular structures are to be undertaken in the near future under the HCM-Network programme entitled "Pre-Normative Research in Support of EC8". For this, simplified global models (storey shear/drift and torsion moment/rotation) are being implemented and calibrated.

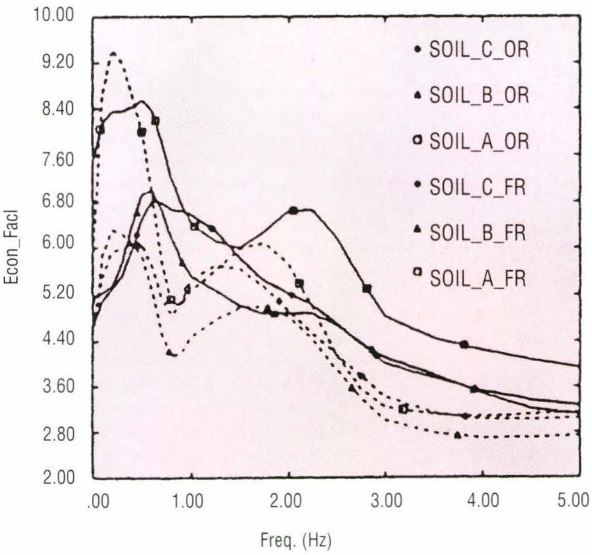

Fig. 6.21 Average economy factor for Friuli-like and Orion-like excitations and different soil conditions (soil A-stiff soil, soil B-medium stiff soil and soil C-soft soil)

The COST-C1 project

COST-C1 is a project intended to study the behaviour of semi-rigid civil engineering structural connections. It is, principally, a framework for R&D cooperation, in the field, allowing for both the coordination of national research, projects and/or the participation of third countries in Community programmes, taking the form of pre-competitive or basic research or activities of public utility. In addition, to the E.C. Member States, six of the EFTA countries are now participating in the project (Austria, Finland, Iceland, Norway, Sweden, Switzerland) as well as Czechoslovakia, Hungary, Poland, Turkey and 'Yugoslavia'.

The Safety Technology Institute participates in this project with two main aims, namely: a) to promote and extend the participation of member states, EFTA countries and other European countries in cooperative research and development activities specially related to ELSA and b) to promote the use of the laboratory by industry for qualification of innovative design concepts and advanced manufacturing technologies.

ELSA participated in the first COST-C1 workshop held in Strasbourg in October 1992, presenting its activities and research programmes [25]. In this meeting it was decided to set-up six working groups in different research areas. A member of the ELSA staff was nominated chairman of the 'Seismic Behaviour/Design' working group whose research programme is now being finalized in collaboration with researchers from several European countries.

The PLEXIS-3C project

The activity on the PLEXIS-3C Project for the finite element modelling of fast transient dynamic problems in structures and fluid-structure systems has continued this year in collaboration with CEA-Saclay. References [26] to [30] should be consulted for other details on this activity.

The main developments in the framework of the collaboration contract with CEA have been the following:
In the area of metal forming models, a fully integrated (2x2) quadrilateral element based on a Godunov algorithm for stress transport has been implemented. The results have been deluding, showing a basic flaw in the formulation. A modified formulation has been devised that could eliminate the problem, but this has still to be implemented and tested.

As far as shell elements are concerned, the 2D conical shell element has been completely revised. A 3D plate element has also been revised and generalized. Work on the 3D degenerated element has only been started, in particular as regards its coupling with fluids.

A critical revision of boundary conditions has been started. This concerns above all the use of linear 'connections' between degrees of freedom based on a Lagrange multiplier algorithm. The possibility to model fluid-structure interactions using such techniques is being explored. The technique has been already applied to the representation of symmetry conditions in general 3D structures, and is being used for the definition of a more general fluid-structure interaction algorithm for ALE sliding situations.

An example is shown in ***Fig. 6.22***: an explosion is supposed to take place in a fluid and the interest is focussed on the effect of the pressure waves on a structure embedded in the fluid itself. Sliding conditions of the ALE type are assumed between the structure and the fluid, while special absorbing elements are placed on the fluid boundary in order to simulate the presence of an infinite media (no pressure wave reflections at the boundaries). A full 3D calculation was performed: a) shows the deformed fluid mesh at various stages, note the growth of the explosion bubble; b) shows the deformed shape of the structure at the same times (clamping conditions are assumed along the two straight edges); c) shows pressure distributions in the fluid, and d) the velocities in the fluid.

In the post-processing domain, a new version of the TPLOT file management and data visualization system has been produced. The software is now fully interfaced with EDF, the Institute's standard data base for signal-like data. The system can now be used also to produce space plots along a curvilinear abscissa. A Motif-based interface has been developed that greatly facilitates the choice of input signals from the data-base.

Finally, a new version of PLEXIS-3C has been produced. This makes available the post-processing interface to CASTEM 2000 recently developed by

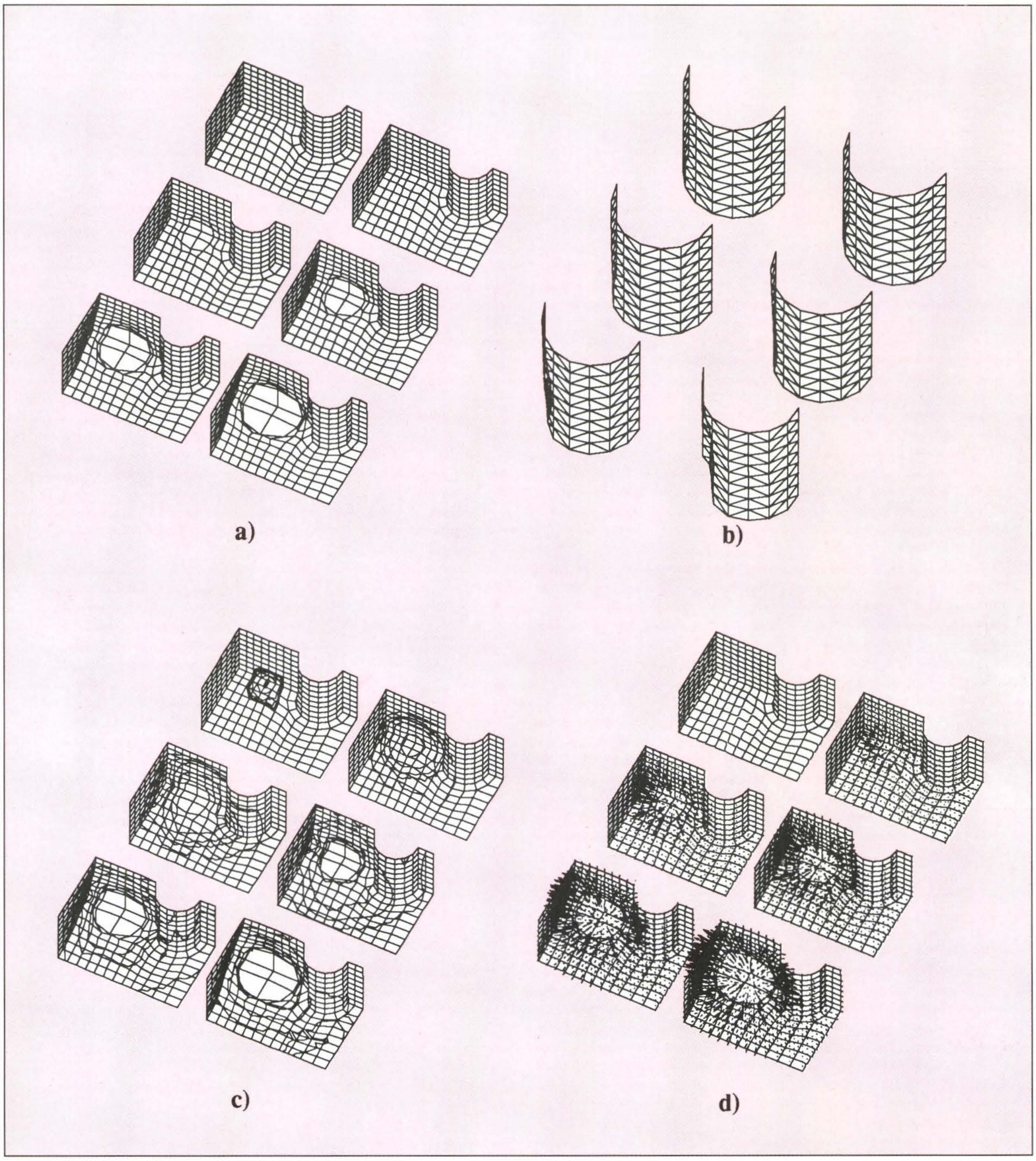

Fig. 6.22 Example of 3-D fluid-structure interaction. An explosion is simulated in a fluid surrounding a thin cylindrical structure. The figure shows a) the deformed fluid mesh, b) the deformed structure mesh, c) the pressure distributions in the fluid and d) the velocities in the fluid at different times

CEA. Other minor advantages are a more F77-compliant source code (many non-standard-F77 features have been eliminated in the programme), the possibility to read lower-case input files, several new models, etc.

References

[1] GUTIERREZ E., MAGONETTE G. VERZELLETTI G. - Experimental Studies on the Loading Rate Effects on Reinforced Concrete Columns - J. Mech. Engrg. Div. ASCE, 1993

[2] KAKALIAGOS A. - Pseudo-dynamic test-setup of a full scale three-storey one-bay steel frame - Technical Note No I.92.103, CEC/JRC, Institute of Safety Technolgy, October 1992

[3] DONEA J., JONES P.M., MAGONETTE G., VERZELLETTI G. - The Pseudo-Dynamic Test Method for Earthquake Engineering - Report EUR 12486 EN, CEC/JRC Institute of Safety Technology, 1990

[4] KAKALIAGOS A. - Behaviour of steel frames with semi-rigid joints under seismic induced loads, Dissertation, Technical University Darmstadt, 1989

[5] KAKALIAGOS A., BOUWKAMP J.G. - Experimental study of steel beam-to-column connections under cyclic loads, submitted for publication in the Earthquake Spectra Journal (E.E.R.I), 1992

[6] KAKALIAGOS A., BOUWKAMP J.G. - Seismic behaviour of a ten-storey steel moment resisting frame with semi-rigid beam-to-column connections, to be presented at the National Earthquake Conference, Memphis Tennessee, 3-5 May 1993

[7] Eurocode 8 (1988) - Structures in seismic regions - Design" Part 1. General and Building, May 1988 Edition, CEC Report, EUR 8849 EN

[8] Eurocode 9 (1984) - Steel Structures-Design-EUR 12266 EN

[9] BUCHET P. - Station Graphique Multi-Ecrans - Technical Note N° I.92.144 - CEC/JRC, Safety Technology Institute, December 1992

[10] PRAKASH V., POWELL G.H. - DRAIN-2DX a general purpose computer program for dynamic analysis of inelastic structures- - University of California, Berkeley, January 1992

[11] SHING P.B., MAHIN S.A. - Pseudodynamic Method For Seismic Testing - UCB/EERC-84/05, 1984

[12] BELLORINI S. - Approccio pseudodinamico all'analisi sismica ed estensione al caso di strutture a massa distribuita CEC/JRC/AMD, ELSA Laboratory - Ispra (VA), Italy, Oct. 92

[13] CALVI M., MAGENES Giovanni e Guido, PAVESE A. - Indagine sperimentale e numerica su un prototipo di edificio in muratura - Dip. Meccanica Strutturale - Università di Pavia, Feb. 92

[14] PINTO A.V., NEGRO P., JONES P.M. - Comparative Analysis of R/C Buildings Designed According to EC8 (Frame and Shear Wall) - Proceedings of the 10th World Conference on Earthquake Engineering, Madrid, 1992

[15] PINTO A.V., NEGRO P. - Dynamic Non-linear Analysis of the 4 Storey R/C Buiding to be Tested in the ELSA-Reaction Wall Facility - Cooperative Research on the Seismic Response of Reinforced Concrete Structures, Interim Report, Lisbon, 1992

[16] NEGRO P. - Ductility, Damage Indicators and Design - Technical Note, Submitted to the Cooperative Research Group on the Seismic Response of Reinforced Concrete Structures, Ispra, Italy, 1992

[17] NEGRO P., JONES P.M., PINTO A.V. - A Reduced Degree of Freedom Approach in the Pseudodynamic Test Method - Proceedings of the 10th World Conference on Earthquake Engineering, Madrid, 1992

[18] PINTO A.V., NEGRO P., JONES P.M. - Comparative Analysis of R/C Building Structures (Frame Shear Wall) Designed According to EC8 - Proceedings of the 10th WCEE, A.A. Balkema, Madrid, 1992

[19] DONEA J., VERZELETTI G., JONES P.M., PINTO A.V. - The European Laboratory for Structural Assessment (ELSA) - Reaction Wall of the JRC - Ispra, Proceedings of the 10th WCEE, A.A. Balkema, Madrid, 1992

[20] PINTO A.V., PEGON P. - Numerical Representaion of Seismic Input Motion - in Experimental and Numerical Mehtods in Earthquake Engineering, Edited by Donea J. and Jones P.M., published by Kluwer Academic Publishers, Netherlands, 1991

[21] PEGON P. - Model implementaion in CASTEM2000: some general considerations and a simple practical realization - Technical Note, to appear

[22] PEGON P. - A Timoshenko simple beam-element in CASTEM2000 - Technical Note, to appear

[23] CASTEM2000 - Guide D'Utilization, CEA/Saclay, France, August 1990

[24] PINTO A.V., JONES P.M. - Analysis of Hybrid Structural Building Structures (Frames/Shear Wall) Subjected To Earthquake Loading - Proceedings of the International Conference on Buildings with Load Bearing Concrete Wall in Seismic Zones, AFPS Edition, Paris, June 13-14, 1991

[25] DONEA J., JONES P.M., PINTO A.V. - The European Laboratory for Structural Assessment (ELSA) - Reaction Wall of the JRC - Ispra, Proceedings of the COST-C1 Workshop, Strasbourg, 28-30 October, 1992

[26] DONEA J., HUERTA A., CASADEI F. - Finite element models for transient dynamic fluid-structure interaction - New Advances in Computational Structural Mechanics, Studies in Applied Mechanics 32, (Ed. Ladeveze P. and Zienkiewicz O.C.), Elsevier, 1992

[27] HUERTA A., CASADEI F., DONEA J. - An arbitrary Lagrangian- Eulerian stress update procedure for coining simulations - 4th Int. Conf. on Numerical Methods in Industrial Forming Processes, NUMIFORM '92, Sophia Antipolis, France, 14-18 Sept. 1992

[28] CASADEI F. - TPLOT an Interactive Data Management System for Transient Problems, 4th Edition - Technical Note N. I.92.26, March 1992

[29] DONEA J., BELYTSCHKO T. - Advances in Computational Mechanics - Nucl. Eng. Des. 134, pp. 1-22, 1992

[30] DONEA J, QUARTAPELLE L. - An introduction to Finite Element Methods for Transient Advection Problems, Comp. Meths. Appl. Mech. Eng. 95, pp. 169-203, 1992

1.7 WORKING ENVIRONMENT

Working Environment is a new research programme which was started up in 1992. At present, the Working Environment at the STI is limited to the subject "Ventilation and Pollutant Transport Modelling" and involves only a small effort.

The general objective which is pursued by this activity is the improvement of working conditions through a more clean, comfortable and healthy environment at the work place, that can be reached with better and more efficient ventilation systems.

The development of accurate and efficient Computational Fluid Dynamics models to deal with turbulent air-pollutant flows is a pre-requisite to fulfill these objectives.

1.7.1 VENTILATION AND POLLUTANT TRANSPORT MODELLING

This project involves the development of a new finite element method for the numerical solution of laminar and/or turbulent incompressible flow problems coupled with the thermal energy equation (TRAFLU computer code series). As well-known, key issues in the development of finite element methods for the numerical solution of incompressible flow problems are the treatment of the incompressibility constraint and the formulation of schemes allowing for an accurate modelling of highly convective situations. In accordance with the mentioned objectives, the following activities have been performed:

- Development of an implicit time-integration scheme for the advection phase based upon a least-squares variational formulation and space-time finite elements. This formulation was developed in the past by the JRC-Ispra for solving 1D transient advection equations and has now been extended to multi-dimensional situations. This space-time least-squares weak formulation can be regarded as a Petrov-Galerkin method in which a modified weighting function appears naturally and requires no "free" parameter. This scheme is shown to be unconditionally stable and second order accurate. The implementation of the multi-dimensional space-time least-squares finite element schemes for the advection equations with source terms in the TRAFLU-2D and TRAFLU-3D codes has been performed with success [1,2]. Reference [2] highlights the ability of UNIGRAPH+2000 to visualise the TRAFLU results in the form of two-dimensional contour plots for both geometrically simple and complex spatial domains.
- The major effort during 1992 has been devoted to the review, the development and the implementation of a turbulence model in the TRAFLU code [3] in order to study ventilation and indoor pollutant transport problems. The finite element algorithms proposed for flows in the laminar regime are in a form which enable incorporation of a two-equation (k-ε) turbulence model easily. However, the convergence of the numerical solutions depends strongly on the nature of the problem treated and on the choice of the appropriate turbulent model and its associated boundary conditions. Hence, each new problem treated should be carefully analyzed to understand its peculiar aspects and additional efforts could be necessary to solve possible new convergence problems.
- The solution of the turbulent natural convection in a 2D-square cavity constitutes a first numerical test case where a low-Reynolds number k-ε model is applied [4]. In this model no wall function formulas are required because the model is valid for the whole flow domain. However, the boundary conditions for k and ε must be specified. The square cavity has a hot left vertical wall (with uniform temperature T_H) and a cold right vertical wall (with uniform temperature T_C). As the test case is aimed at interpreting natural convection in the turbulent regime, the Raleigh number, Ra, should be at least 10^{10} for air (with the Prandtl number Pr=0.71). Calculations were performed for Ra=5×10^{10} on 20X20 meshes. The temperature difference between the two side walls was 5°C. The top and bottom wall were adiabatic. ***Figs. 7.1.a*** and ***7.1.b*** show calculated streamline and temperature contours. The flow consists of a large clockwise vortex, and from the contours of the stream functions, it can be seen that there are two small clockwise vortices near the mid plane.

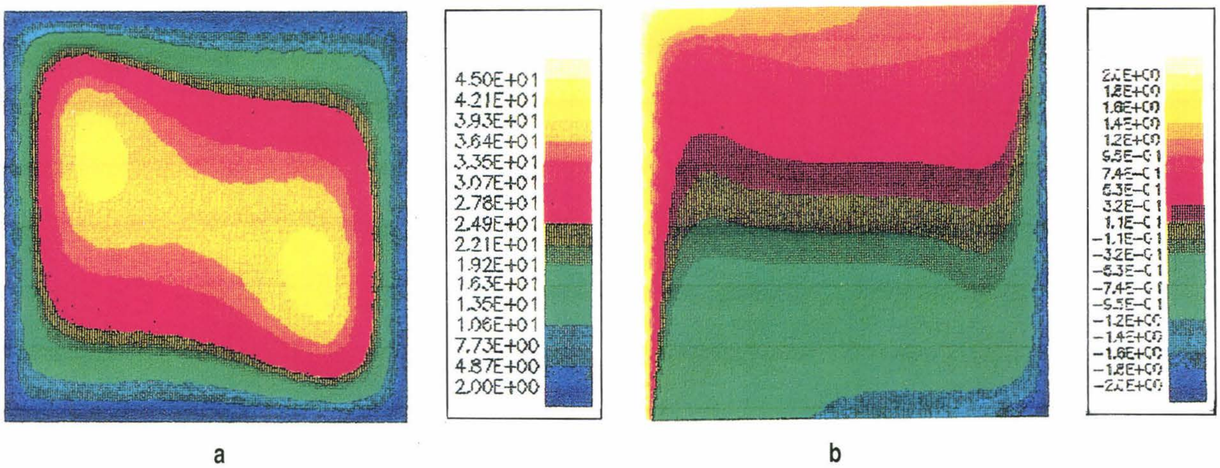

Fig. 7.1 Streamline contours (a) and temperature contours (b)

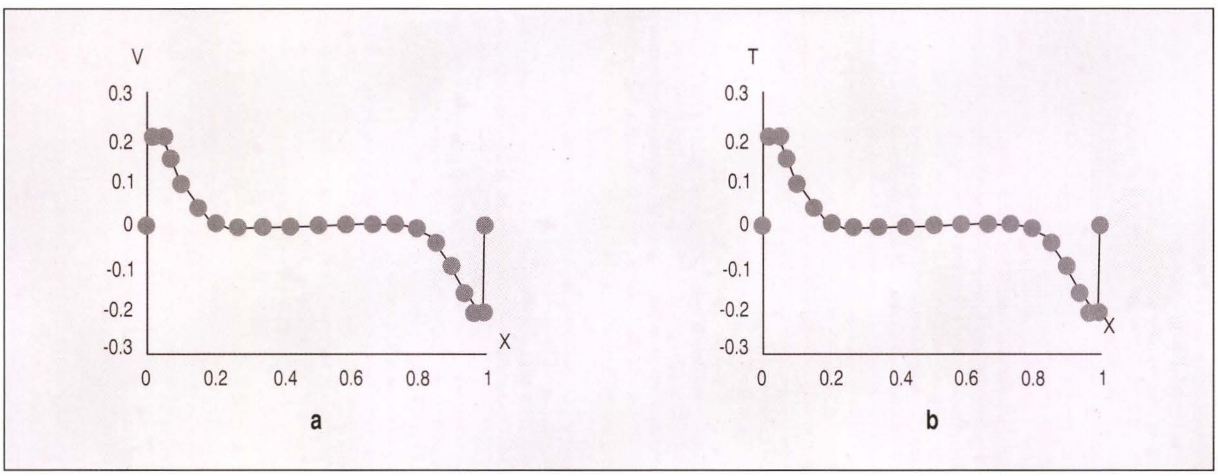

Fig. 7.2 Vertical velocity (a) and temperature (b) at the centre of cavity (y=L/2)

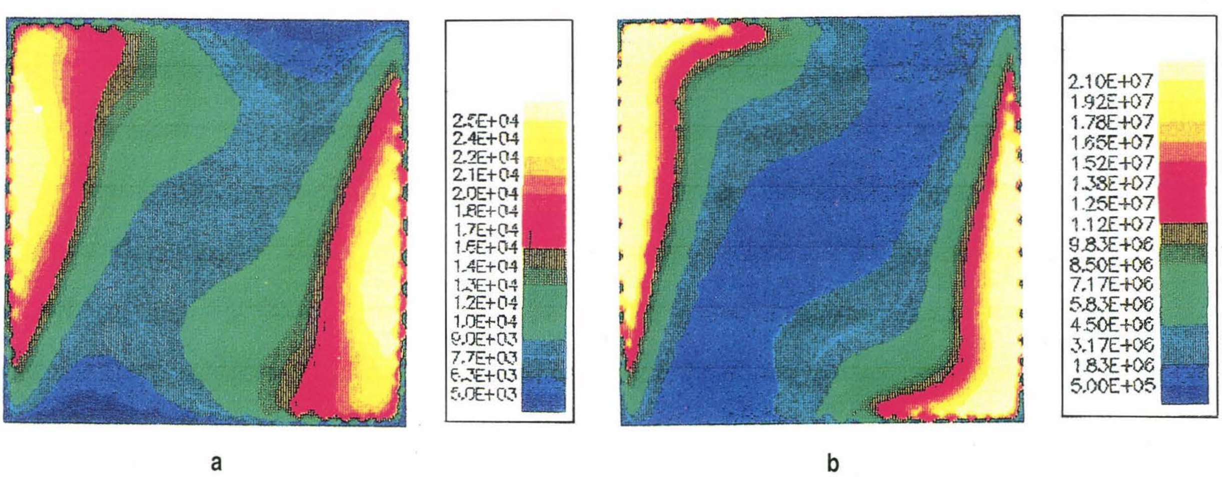

Fig. 7.3 Turbulent kinetic energy contours (a) and dissipation rate contours (b)

Fig. 7.2 shows the vertical velocity profile (***7.2.a***) and the temperature profile (***7.2.b***) at mid plane. It can be noted that ***Fig 7.2.a*** justifies the need for non-uniform grids which allow several discretization points in the dynamic boundary layer where the maximum of the vertical velocity is observed. The vertical velocity is equal to zero in the greater part of the cavity. The falling of the vertical temperature gradient at the centre of cavity - ***Fig. 7.2.b*** - means that the central area is nearly isothermal for high Raleigh numbers.

The turbulent kinetic energy and dissipation rate contours are plotted in ***Fig. 7.3***. We can see that the behaviour of the fluid is turbulent in the top left and bottom right sections of the cavity. These results are in good agreement with other computer code calculations performed in the framework of international benchmark exercises. Next steps in the TRAFLU development and validation will be the calculations of forced convection turbulent air flows representing ventilation air motions. Model improvements foreseen in 1993 will be focused on a gaseous pollutant transport formulation and on discrete representation of boundary conditions.

References

[1] NGUYEN H., PETROV-GALERKIN A. - Formulation based on the least-squares finite element concept for multi-dimensional advection-diffusion equation - Application to natural convection problems, Technical Note N°I.92.25 - JRC-Ispra, March 1992

[2] AL-KHUDHAIRY D.H.A. - Application of the explicit Taylor-Galerkin and implicit Petrov-Galerkin schemes to two-dimensional fluid flow problems, Technical Note N° I.92.46 - JRC-Ispra, May 1992

[3] NGUYEN H. - Turbulence models for practical applications: A review, Technical Note N° I.92.67 - JRC-Ispra, June 1992

[4] NGUYEN H., PETROV-GALERKIN A. - Least-squares finite element algorithm for prediction of room air motion, paper submitted for presentation at INDOOR AIR - Helsinki-Finland, 4-8 July 1993

2 SUPPORT TO COMMUNITY POLICIES

2.1 SAFEGUARDS

The Safeguards Support Programme aims at assisting the Euratom Safeguards Directorate (ESD, DG XVII, Luxembourg) and the IAEA in the implementation of safeguards instrumentation required in the frame of both the Euratom and the Non Proliferation Treaties for Special Nuclear Material. The objectives of this work are:

- *to develop reliable non destructive assay instruments, associated data acquisition systems, interpretation models and software for user friendly applications of the various techniques*
- *to provide after sale services and training to both inspectorates*

The support to the Commission activities by the STI are subdivided in two parts:

- *Support to ESD*
- *Support to IAEA (through DG I)*

Both support programmes are structured following specific task sheets describing the activities pertinent to each task and the relevant programme achieved. The subjects of ESD and IAEA supporting activities are listed in ***Tables 8.1*** *to* ***8.3****. Only those for which significant progress have been achieved during 1992 are illustrated here.*

2.1.1 SUPPORT TO ESD

Training

In addition to the previous courses (***Fig. 1.1***), two new training courses have been prepared in 1992 concerning statistics for safeguards and basic concepts in NDA physics. The complete "menu" of courses held by the STI is reported in ***Table 8.1*** and ***Fig. 8.2***.

Table 8.1 PERLA training "MENU"

TRG-1	Verification of uranium enrichment by gamma ray spectroscopy
TRG-2	Determination of plutonium isotopic composition by gamma ray spectroscopy
TRG-3	Passive neutron measurements
TRG-4	Active neutron measurements
TRG-5	Physical inventory verification (PIV) at an HEU facility
TRG-6	PIV at a Pu(MOX)facility
TRG-8	PHONID measurement of 235U mass
TRG-9	Basic physics for non destructive assay of nuclear material
TRG-10	Introductory applied statistics for safeguards

During 1992, fifteen training courses have been held as follows: Each of TRG-1, TRG-3 and TRG-8 have been held on three occasions (total 9); TRG-2 has been given twice and each of TRG-4, TRG-5, TRG-6 and TRG-10 has been given once.

TRG-6 : MOX PIV training course

A MOX PIV training course for ESD was carried out at Luxembourg and Ispra in the period 2nd-30th October 1992. The course covered:

- practical experience in the two major Euratom MOX facilities;
- methodology for verification planning, material balance (MUF) and material balance uncertainty, application of D-statistics;
- MOX PIV verification exercise in PERLA involving standard neutron and gamma techniques and data evaluation exercises.

TRG-9: Basic physics for non-destructive analysis of nuclear material

The subjects of the course were:

- Atoms and nuclei
- Radioactivity, types of radiation

Fig. 8.1 ESD inspectors at a physical inventory verification (PIV) training course

- Nuclear reactions
- Interaction of radiation with matter
- Attenuation process
- Radiation detection methods
- Properties of measurement methods and applications
- Measurement aims and measurement errors

The course was integrated with extensive demonstrations and experimental exercises.

TRG-10: Introductory applied statistics for safeguards purposes

The course is intended to introduce safeguards inspectors to statistical methods in general but with a particular focus on NDA measurement data treatment. It introduces statistical concepts by presenting them as the response to problems involving decision-making with measurement data. It provides an elementary introduction to:

- distribution theory (Gaussian, t, Chi-squared, Poisson)

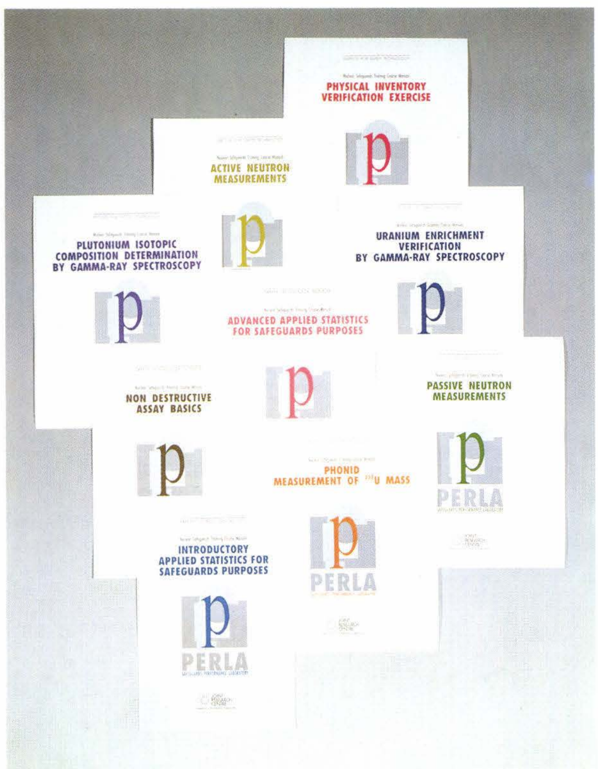

Fig. 8.2 PERLA training menu

- estimation and confidence intervals
- error propagation
- calibration curve fitting
- hypothesis testing

Support to ESD in the field of NDA and instrument development

Active tasks in progress are listed in *Table 8.2*.

Table 8.2 List of ESD active tasks in progress in December 1992

NDA-3.4	Construction of a PHONID 3bis
NDA-6.3	Consultancy in neutron physics
NDA-7.3	Consultancy in gamma spectrometry and technical assistance
NDA-7.4	Upgrading of enrichment meter softwares
NDA-10	Improvement of MTR gamma-scanners
NDA-16	Software for active and passive neutron correlation instruments
NDA-17	Data-base for DA/NDA measurements
NDA-19	Calorimetry
NDA-20	Neutron and gamma software support-unattended systems
NDA-21	Pu waste measurement campaign in Siemens Bw-MOX
NMA-6	Application of the D-statistics

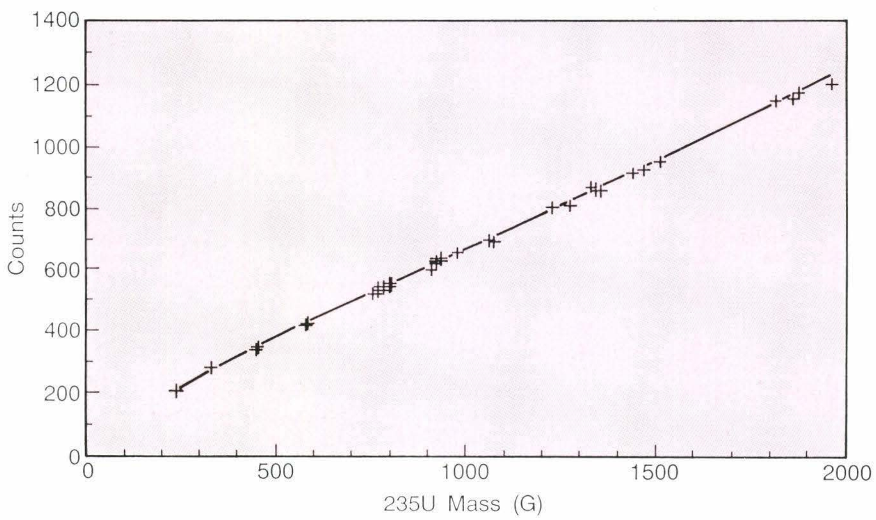

Fig. 8.3 Count rate of PHONID 3b for pellets in cylindrical containers as a funciton of ^{235}U mass in grams

NDA-3.4: PHONID 3b

One PHONID 3b device has been installed for permanent use at BNFL Springfields (GB). During 1992 the NFC Unit of STI has given a full support for the PIV measuring campaigns at ENUSA (E), FBFC (B), BNFL (GB) and Siemens (G), entailing installation of the mobile device, preparation of Sb source capsules, loading-unloading Sb-sources, and calibration of devices.

In addition to the normal PIV support, a special measuring campaign has been performed in collaboration with ESD (DG XVII, Luxembourg) and RBU Siemens to study the accuracy of the PHONID 3b device for measurements of samples containing U-pellets in cylindrical containers. A first analysis shows that such samples can be measured with an accuracy of about 1.5% using the adapted calibration function:

$$C = a\ m\ (1+b\ e^{-cm})$$

with C being the count rate, m the ^{235}U mass of the sample and a,b,c fitting constants. The measured count rates together with the calculated ones (full line) are shown in *Fig. 8.3*. For the first time in 1992 training courses (in total three) have been held on Active Neutron Interrogation using the PHONID device. Six ESD inspectors attended each course.

NDA-7.3: Consultancy in gamma spectrometry and technical assistance

Under this general task a number of activities have been carried out:

- A new generation of miniature multichannel analyzers is coming on the market: it is the task of STI to test those new products in support to ESD, to show their performances and possible uses for safeguards inspections. Amongst others, PERLA staff has tested a new ORTEC analyzer system with satisfactory results.
- A combined neutron/gamma assay system

Fig. 8.4 The Siemens/ESD/IAEA PuO_2 input measurement system at PERLA

for the measurement of PuO_2 input at a MOX fuel fabrication plant was extensively tested (3 week period) in the new PERLA facility, in the presence of ESD and IAEA inspectors (***Fig. 8.4***).

The system, called PIMS (Plutonium Input Measuring Station), consists of:

- a Ge detector of new design
- a passive neutron counter

The participants to the test of the equipment were from LLNL (USA), CANBERRA (B,G), JOMAR (USA), SIEMENS (G), ESD and STI.

The Ge-detector has been designed by LLNL and constructed by CANBERRA. Since it has two different sensitive sections, it can collect and process signals from lower and higher spectrum energy regions. For this reason it has been called telescope detector. The tests performed at PERLA gave very satisfactory results.

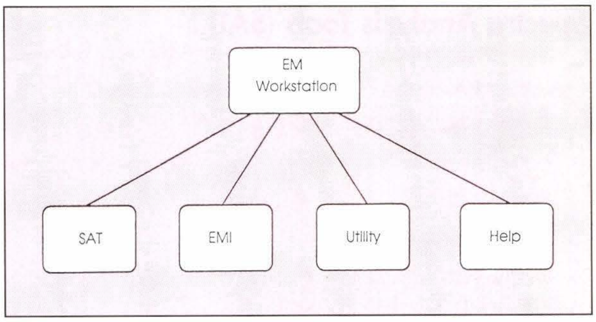

Fig. 8.5 First layer of the EM% software

NDA-7.4: Enrichment meter

The enrichment meter (EM) was developed as an NDA gamma spectrometry workstation. This upgraded version is conceived as a "customer oriented" tool which provides full support to the operator during all measurement phases, starting from the instrument set-up and prompting him through the programme till to the final report of the inspection campaign. The most prominent features of the new system are:

- high accuracy (use of a high resolution HPGe detector) for all kinds of samples including freshly separated fuel;
- complete data handling including comparison of different measurement and/or inspection files;
- possibility of preselecting the desired accuracy.

Software structure

The EM-Workstation software has a layered structure. The first layer contains the main environments of the system: SAT, EMI, Utility and Help. The SAT (Spectrum Analysis Tools) environment allows to visualise the spectra stored in the hard disk. The EMI environment groups all function to perform an inspection such as sample identification and inspection, data acquisition and analysis, comparison of declared and computed data. The "Utility" menu collects the procedures to download and to restore inspection data from a floppy disk. "Help"calls on the line help manual. The first layer of the EM Workstation is schematically represented in *Fig. 8.5*.

Each of the EM-blocks is made up by one or more layers. They are briefly described in the following section on SAT and EMI structures only.

2.2 EXPERT ANALYSIS IN SUPPORT OF DG XXI

This activity deals with the evaluation of scientific instruments imported from the non-community countries. Provided that they are employed for scientific research, have a high scientific value and, of course, no equivalent alternatives available in the Communities, such instruments are exempt from custom duty. The dossiers to be analysed concern imported instruments for which the customs of the Communities Countries have refused the above tax exemption.

To this end the STI of the JRC-Ispra has made available during 1992 a technical support and has participated to meetings held by the Custom Duty Free Committee in which controversial cases were discussed. In case of controversy between the countries, the scientific opinion expressed by the STI expert team is a determining factor for the decision of the Committee. The decisions of the Committee can be contested at the Court of Justice of Luxembourg. In such a case a scientific support has to be supplied by the STI expert team to the Juridical Services of the Commission.

Because of the closure of the Amdahl services (on June, 30th 1992), the RAPID data-bank (and the relevant management and statistical codes) was converted into the actual version for a Personal Computer.

2.3 EXPLOITATION OF NEW INSTRUMENTATION VALUE PROGRAMME - DEVELOPMENTS

2.3.1 TK FLOW MEASUREMENT-SYSTEM

The TK flow measurement system was developed to measure the flow velocity in thermal hydraulic installations (like LOBI) where available flow meters fail. The TK-system has been successfully used for the LOBI data analyses. In 1992 a license was given to a German company (Löffel Verfahrenstechnik, Karlsruhe). The aim of the project in 1993 is to give support to this company to market the instrument.

Physical principle

Small temperature fluctuations are inherent to flows in thermal hydraulic systems. These random temperature fluctuations are detected by two temperature probes in the flow. The distance between the probes in the flow direction causes a time delay between the two signals, which can be measured by correlation analysis. The flow velocity can be calculated directly from the time delay and the distance between the probes.

Technical description

The TK-instrument consists of a probe head (with two thermocouples placed at a known distance) inserted into a flow channel. A two-channel amplifier (with high-gain, low noise electronics and automatic gain control) is used to improve the temperature fluctuation signals. Calculation of the correlation function is performed either by a hardware correlator (off-line multi-channel analysis) or by PC-based software (on-line instrument). The resulting velocity data are validated using specifically developed software. A typical output from the on-line version is shown in *Fig. 8.9*.

Advances

The TK-instrument was licensed to Löffel Verfahrenstechnik. A new amplifier for the temperature fluctuation signals was developed and successfully tested. The final system was presented integrated in a demonstration loop at the INTERKAMA, Düsseldorf in October 1992.

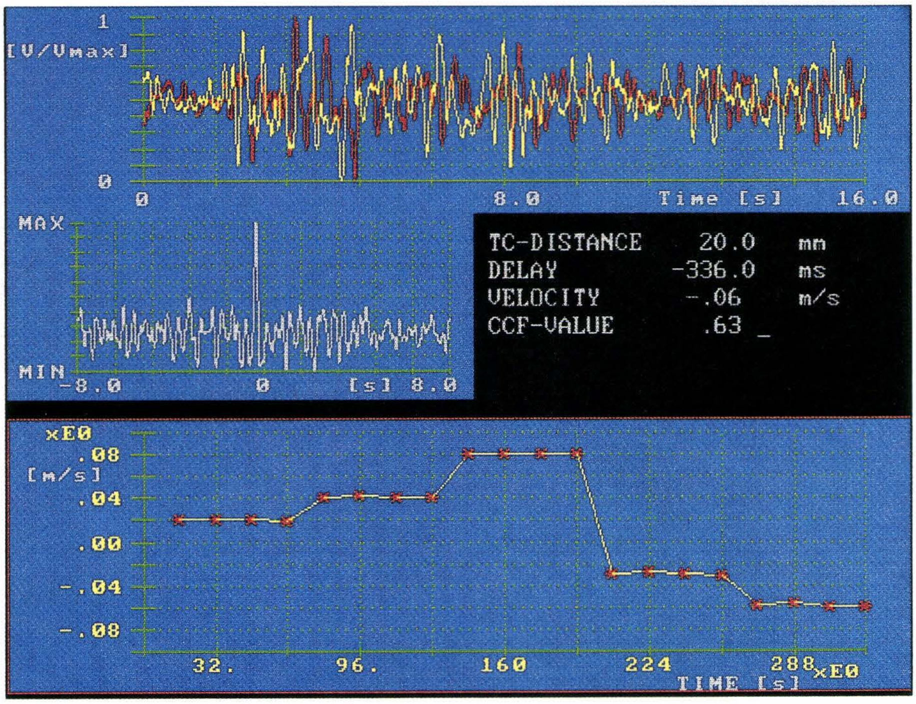

Fig. 8.9 Video display of the on-line version for demonstration experiment; top: Normalised signals from the 2 thermocouples used to calculate the cross-correlation function; centre: Cross-correlation function; peak represents time delay; bottom: Velocity as function of time

2.4 PARALLELISATION OF LARGE COMPUTER CODES - SUPPORT TO DG XIII

The Safety Technology Institute has a number of large codes for mathematical modelling of physical phenomena.

The codes have been under development and/or modification during several years. A large number of programmers have worked on parts of these codes. Various types of numerical methods are applied. The programming language is exclusively FORTRAN in all its dialects. The extended maintenance of the programmes has resulted in the introduction of some quite "tricky coding".

In order to apply the existing models on larger or more complex problems the run time must be reduced. As the codes can require weeks of CPU time for certain parameter studies, a performance increase of an order of magnitude would already help many users.

Rather than rewriting all the codes, it has been decided to port existing codes to run on fast hardware configurations such as parallel computers. This will give a valuable experience in the field of parallel computing which can be exploited when new programmes are developed.

Portability of the codes is an important issue so the porting to the parallel architecture should be performed with a minimum of modifications to the programmes. This approach will, at least in the beginning, lead to a coarse grain parallelisation (i.e. large lumps of programme on each processor).

The parallel computing equipment used at the Safety Technology Institute is a Quintek Fast9/9T Transputer Board with 9 T825. In addition, an existing networked array of UNIX workstations is applied as a virtual parallel machine. The network communication is based on Berkeley sockets and Network Computing System (NCS).

Three codes have been selected for the first phase of the parallelisation project. These are VICTORIA, ISPRAMIX and PLEXIS-3C.

- The VICTORIA programme is for safety analysis of nuclear power plants [1]. It calculates the physical and chemical interactions within the reactor circuit of fission products released as a consequence of a hypothetical accident.
- The ISPRAMIX is a code for the modelling of hydraulic aspects of oceans and seas. It is based on finite volume methods [1]. The algorithms are relatively small but repeated many times for an enormous number of volume elements.
- The PLEXIS-3C is further descibed below.

The parallelisation of the VICTORIA code was reported in 1991 [2]. The ISPRAMIX has only recently been "released" for parallelisation so the major work of 1992 has concentrated on the PLEXIS-3C [3,4].

2.4.1 THE PLEXIS-3C

The PLEXIS-3C programme is a result of a joint project between the CEA (France) and the JRC, Ispra. It is a computer programme for the analysis of fast transient dynamic phenomena in structures. Compressible fluids can be treated by the programme. Non-linearity of material (plasticity, viscoplasticity) and of geometry (displacement, rotation and strains) are accounted for. Fluid-structure interaction phenomena are also modelled. The model is based on the finite element spatial discretisation of the equilibrium equations and assumes a Lagrangian representation, i.e. the finite element mesh follows the motion of the material particles.

From the very beginning the PLEXIS-3C has been conceived and designed as a general purpose programme for the treatment of transient dynamic problems. Being a joint project, PLEXIS-3C has been designed to facilitate developments performed by different groups simultaneously. The main features to allow such a development has been orthogonal modules (almost no interaction between the various modules) and an unified data structure in which the individual users can only access the necessary elements. The programme is written in FORTRAN and contains several myriad's of statements. Various programming styles have been applied. Certain parts of the programme are still based on the '66 standard and require special treatment.

To ease the development of the parallel version of the PLEXIS-3C a reduced version has been taken for the working reference. This version of the programme contains all the necessary informatics features of the PLEXIS-3C and is only reduced in terms of applicability for the problem description. Many of the optional models which are not requested by the test input data have been scraped off.

The resulting code is of some 20,000 FORTRAN statements. The problem size of the data region has been defined to allow for the use of the relatively limited memory on the parallel hardware. About one Megabyte of memory is used for the data region. The original layout of this region has been maintained so that the methods applied can be scaled.

The code consists of two logically distinct parts:

- The problem specification and initialisation: this is the major part of the programme counted on number of statements.
- The solver, which has relatively few statements, but which is spending almost all the CPU time for large problems.

The PLEXIS-3C has been physically split into these two parts. The large first part is turned into a pre-processor for setting up the problem. This part can be fitted neither on the PC nor on the Transputer array. It is executed on a UNIX workstation and the resulting set-up file with the problem specification is transmitted to the PC hosting the Transputer array. The smaller second part which performs the CPU intensive calculations is executed on the parallel machine.

The lay-out of PLEXIS-3C suggests a master/slave structure where the sequential part of the code is residing on a host machine (master) and a number of processor nodes (slaves) calculate the parallel part. However, it turned out that the data structure of PLEXIS-3C lead to a too large communication overhead.

To run the parallel version of PLEXIS-3C on the Transputer array the CUBIX model of the Parasoft Express System was chosen. A version of the programme is loaded on each individual processor and run in parallel. The main do-loop for the element calculation has its limits specified according to the processor where it resides. After each termination of the main loop the results are broadcast to all the active transputers. Thus, the resulting programme is a compound code which is slightly sequential and heavily parallel.

An advantage of the CUBIX system is the limited number of systems functions needed. These functions are included as stubs on the UNIX machine. Thus, the code travels without any modification between the UNIX workstation and the transputer array.

For the implementation the major problems were caused by the lack of a real operating system for the transputers and the I/O from the code. A distributed and extended use of output functions and error message functions in the PLEXIS-3C caused synchronisation problems. The output had to be send through a few controlled channels in order not to jam the system.

Table 8.6 shows the speed-up as a function of the number of processors. This speed-up is not linear as a significant part of the programme is sequential. However, if only the parallel part of the programme is considered, an almost linear speed-up is obtained. In this reference case the number of elements was only 120. This caused a relatively large communication overhead when the number of nodes increased. The problem should vanish when dealing with real large cases with thousands of elements.

The actual version of PLEXIS-3C has some 30% sequential and 70% parallel code. It is expected that

Table 8.6 The performance of PLEXIS-3C as function of the number of processor nodes applied

Number of Nodes	CPU Total	CPU Parallel	Transmission
1	74.9 sec	51 sec	0.00 sec
2	49.4 sec	25 sec	0.36 sec
3	39.7 sec	16 sec	2.10 sec
4	36.8 sec	13 sec	1.87 sec
5	34.8 sec	11 sec	2.09 sec
6	32.6 sec	9 sec	2.04 sec

3.1 EUROPEAN ASSOCIATION OF STRUCTURAL MECHANICS LABORATORIES (EASML)

The EASML is an association intended to promote European collaboration in the fields of structural mechanics and earthquake engineering. Set-up in 1989 on the initiative of the Institute for Safety Technology, the Association continues to grow steadily and an effective cooperation between its members and the ELSA laboratory is now well under way.

The building Research Establishment and Imperial College of the U.K., as well as the University of Oviedo, Spain, the Politecnico di Milano and RWTH Aachen joined the Association in 1992, bringing the total number of associated laboratories to 29.

Most members of the Association were present on the occasion of the official opening ceremony for the ELSA reaction-wall facility on October 16, 1992, at Ispra. This event actually attracted over 300 attendees from all over the European Community. Dr. A. RAVARA from L.N.E.C., Lisbon, a founding member of the Association, highlighted in his talk the new opportunities in the Community for collaboration in the fields of earthquake engineering and structural dynamics offered by the European Commission.

The inauguration of ELSA also provided an opportunity to review, during an afternoon workshop, the progress achieved in the execution of the collaborative research programmes of the Association. The new possibilities arising through the launching of the "Human Capital and Mobility" programme of the Commission were also discussed.

Four working groups were established within the Association to study reinforced-concrete, steel/concrete composite and masonry structures, as well as to promote the further development of testing techniques, in particular the pseudo-dynamic test method used at ELSA.

As regards the reinforced-concrete group, the second phase of its programme (see STI Annual Report 1991) is now approaching completion. In this phase experimental aspects are being emphasized with specific research topics on the definition of damage indicators and failure criteria in plastic hinge regions of reinforced-concrete structures subjected to seismic loading. Included in this programme is the final design and non-linear analysis of a full-scale four-storey frame structure which will be tested in the ELSA facility in 1993.

The work of the group studying steel/concrete composite structures is also progressing. Major aims are to investigate the reliability of the pseudo-dynamic tests in capturing the earthquake response of composite structures, to get experimental evidence on the dynamic behaviour of beam-column composite connections and to investigate the ability of existing numerical models to capture this behaviour. The results of this group are expected to lead to a full-scale test on a 3-storey composite frame to be performed in the ELSA facility around 1994-95.

The programme of work for the group on masonry structures has still to be finalized as a function of the results of current research in the associated laboratories, in particular at Pavia and Lisbon.

Work of the group on test method development consists of numerical and experimental studies in support of the validation and further improvement of the pseudo-dynamic method. In particular, an assessment is being made of the effects of errors due to instrumentation, control strategy and integration algorithms.

New possibilities for the Association have arisen through the "Human Capital and Mobility" (HCM) programme launched in mid 1992 by the European Commission (DG XII).

In response to the HCM call for proposals of June 1992, the setting-up of a cooperation network involving 19 laboratories from the Association has been proposed with the aim of performing pre-normative research in support of Eurocode 8 for the design of structures in seismic areas. This proposal has been accepted in November 1992 and the network activity is expected to start in March 1993 after signature of the related contract with DG XII.

On the same occasion, proposals were submitted for the inclusion of 5 experimental facilities from the Association, of which ELSA makes part in the "Access to Large Installations" chapter of the HCM programme. These proposals have also been accepted and, together with the approved cooperation network, they should provide further support and open up new opportunities for strengthening the collaboration links within the Association.

4 EXPLORATORY RESEARCH

- 4.1 Studies of accelerator based transmutation systems
- 4.2 Boron neutron capture therapy studies
- 4.3 Development and qualification of crucibles for high temperature melts
- 4.4 NDA detection limits for plutonium in radioactive waste
- 4.5 Iron ore reduction
- 4.6 Purification of the gaseous feed stream to fuel cell
- 4.7 Neural networks for chemical reactor control
- 4.8 Application of chaos theory to batch and semibatch chemical processes
- 4.9 Study of the true stress-strain response of plain concrete by bundle Hopkinson's bar technique

4.2 BORON NEUTRON CAPTURE THERAPY STUDIES

There is a rapidly growing interest in Boron Neutron Capture Therapy (BNCT) after the encouraging results obtained by H. Hatanaka using a mixed surgical and radiative treatment of tumours.

Progress in pharmacokinetic boron compounds and the development of suitable neutron sources makes BNCT a promissing candidate in the treatment of certain types of cancer. A Concerted CEC Action started in 1989 with the following objectives:

- Installation of a plant for BNCT at the JRC Petten using a channel of the HFR Reactor;
- Demonstrating the feasibility of the plant, and after the necessary controls, running the facility for medical treatment of brain tumours;
- Support to the other European projects on BNCT, in particular the development of accelerator driven neutron sources.

Progress up to the end of 1992

The JRC Ispra is participating in the Concerted Action Group and its work is centred around the theoretical analysis of the proposed irradiation plant predicting neutron beam performance and dose-rates. Recent versions of codes for radiation transport and nuclear libraries were set up with particular emphasis on:

- DOT and DORT [two-dimensional S-N transport codes]
- TORT [three-dimensional S-N transport code]
- NJOY-89 [cross sections retrieval and preparation code]
- MCNP [monte-carlo transport code]

The libraries are based on files from the ENDF-B4, ENDF-B66 and JEF-2 master tapes. These tools were used for preliminary studies of the Petten plant [4] and of a similar plant [5] to be installed at TRIGA reactors using a converter plate made of U_{235} to reinforce the neutron source. During 1992 the activity was devoted to an analysis aiming at an improvement of the BNCT facility [6]. This device, already installed and tested, shows a good agreement with the performance characteristics established by the project team. It seems, however, that the therapy requirements could be improved if fast neutrons contamination could be reduced and the level of total flux could be increased by changing the present design of the filter inserted in the HFR channel. The calculations cover the four months period, February-May 1992.

A second cycle of calculations, planned for checking and refining the more important findings, will be resumed as soon as the transfer of codes and data from the main frame to workstations will be completed.

The changes in the layout concern the substitution of liquid Argon in the filter with a solid material (TiD_2 or Al_2O_3), serving as neutron moderator and gamma shield. The insertion of an Al shield at the channel entrance and the use of Bi as channel wall cladding was studied. The results have shown that the liquid Argon shield should be kept, because the other materials (TiD_2/Al_2O_3) are less effective considering the geometric constraints of the channel. Only the supplementary Al shield offers possibilities of improvement. The best solution which was found requires:

- a supplementary Al shield of 16.7 cm thickness
- a liquid Argon shield of 126 cm thickness
- a solid filter placed halfway in the channel composed of:
  - Al 12 cm
  - Ti 0.1cm
  - Cd 0.016 cm

For this arrangement the neutron flux at the therapy position becomes: 8.4×10^8 n/cm^{2}sec (target value 1×10^9 n/cm^{2}sec) and the neutron mean energy 7.6 keV (target value 5.0 keV). *Table 9.2* gives the values for the cases which are near the optimum.

Table 9.2 Values for near optimum cases

Case Number	Supplementary Shield Aluminum Thickness cm	Filter				Neutron Flux n/cm²sec	Mean Neutron Energy keV
		Al Thick cm	Ti Thick cm	Cd Thick cm	Ar Thick cm		
1	16.7	12.0	0.053	0.016	100	$1.14X10^9$	8.49
2	16.7	12.0	0.053	0.016	110	$1.04X10^9$	8.09
3	16.7	12.0	0.053	0.016	118	$9.68X10^8$	7.80
4	16.7	12.0	0.10	0.016	118	$9.01X10^8$	7.90
5	16.7	12.0	0.10	0.016	126	$8.35X10^8$	7.60
6	16.7	12.0	–	0.032	126	$9.80X10^8$	8.41
7	16.7	12.0	–	0.032	148	$8.50X10^8$	7.95

4.3 DEVELOPMENT AND QUALIFICATION OF CRUCIBLES FOR HIGH TEMPERATURE MELTS

A crucial problem in many melting processes is the availability of high temperature crucibles for large charges of metallic materials. These crucibles have to resist temperatures up to 2000°C for times of a few hours and due to high cost they have to be used a number of times. The solution of this problem is essential for extending the capability of the FARO and KROTOS facilities used for the study of severe accidents in LWRs as well as for processes related to waste treatment.

Tests are planned in FARO with large amounts of "corium" melts. In addition to the UO_2 and ZrO_2 ceramics, metallic composites (stainless steel, Zr) are included in the corium. The problem to be solved is the melting of up to 200 kg of these metals in crucibles which have to resist 2000°C. In KROTOS the phenomena of steam explosions using simulant materials are successfully investigated. To ensure the continuation of the test programme crucibles have to be developed, for melting realistic oxide or metallic core materials at 3000°C or 2000°C, respectively.

In the field of nuclear waste treatment, the detritiation and compaction of solid wastes before their storage and disposal is problematic, concerning in particular the waste coming from exhausted boron carbide (B_4C) control rods used in BWRs. A possible process is first to chemically decompose the B_4C to ensure the complete removal of tritium and then to melt the eutectic alloy formed by B_4C, SS and structural materials. This process needs special crucibles for iron-based melts at 1700°C.

Progress up to the end of 1992

During 1992 contracts with industrial companies were taken for the procurement of adequate materials for crucible fabrication. As a result, three candidate crucible materials which should resist SS/Zr melts at 2000°C were identified for further compatibility studies. In these compatibility tests SS/Zr charges of some tens of kg were inductively melted to 2000°C in small crucibles of ZrO_2 and MgO ceramics of various densities.

High density ZrO2 (5.5 g/cm^3) showed excellent chemical compatibility with the metallic melt in argon atmosphere, but the crucible cracked by using it for remelting. This problem of thermal fracture may be overcome by applying composite crucibles of the same material for large mass melting. The results of numerous compatibility tests encouraged the developer to purchase of ZrO_2 crucibles of 30cm diameter and height, of both monolithic and composite design to be used for melt masses of 50 kg.

For this purpose a design study was performed to convert an existing radiation furnace into an induction furnace. A medium frequency (3-5 kHz) power supply of 250 kW, available at JRC Ispra, will be connected to this furnace. Detailed designs of the furnace components, e.g. the matching MF-coil and current connections are in progress. As an intermediate step for the target melting plant of 200 kg charges, it will be attempted to achieve in this modified furnace the melting of 50 kg SS/B_4C and SS/Zr mixtures at 1700-2000°C and to study the melt release through the bottom plate of the crucible.

Future KROTOS tests with prototypic UO_2/ZrO_2 core melts need adequate crucibles. For the reason of material compatibility at nearly 3000°C the only possible crucible material is tungsten. In the KROTOS test procedure the crucible (containing 5 kg of melt) is dropped onto a puncher which ruptures the crucible bottom, thus resulting in consecutive melt draining. Therefore, W crucibles were needed with 0.2 mm thin bottoms, which could not be offered by companies, specialised in W-manufactoring. However, the JRC central workshop after many attempts succeeded in preparing crucibles to the specifications using the electroerosion technique. Also due to this successful development a contract between USNRC and JRC has been concluded through which the execution of a UO_2/ZrO_2 test series in KROTOS will be financed for 1993.

4 NDA DETECTION LIMITS FOR PLUTONIUM IN RADIOACTIVE WASTE

A powerful tool for determining the spontaneous fission rate and the mass number of Pu isotopes is the neutron signal correlation method. By this method it is possible to introduce, in the measurement of Pu contaminated waste barrels, corrections for the (α-n)-reaction rate of PuO_2 and for the particular detection probability of the Pu debris dispersed in unknown locations of the barrel. With low Pu contamination, i.e. below 2g of Pu in a 220 dm^3 waste barrel, the neutron correlation technique is disturbed by cosmic radiation producing signal bursts with a high neutron multiplicity. These neutron bursts cannot be considered to be statistically stationary within the time scale of the measurement of a waste barrel. To reduce the influence of cosmic radiation it is necessary:

- to optimize the neutron detector head so that the optimal neutron detection probability can be reached,
- to filter the neutron signal pulse train so that the less frequent signal bursts with high neutron multiplicity can be strongly reduced and the spontaneous fission neutrons, lost in the filter, can be corrected by theoretical means,
- to provide shielding for stopping the μ-mesons in heavy metal and the neutrons in polyethylene shielding,
- to surround the neutron detector head by a liquid scintillator which detect passing neutrons and stop any signal correlation measurement for a short duration after each neutron passage.

The first two options have been applied in the experimental work described below. The detector head used for the experiments, the results of measurements using an updating dead time filter and the proposed modifications of the detector head to obtain the maximnm neutron detection probability are illustrated hereafter.

Progress up to the end of 1992

The waste barrel monitoring assembly

The waste barrel monitoring assembly consists of three different elements: the detector head, the electronic chains, and the data acquisition and analysis system (*Fig. 9.1*). The detector head has a cylindrical cavity with a horizontal axis housing a waste barrel up to the size of 220 dm^3. 60 ^3He neutron detectors are arranged in a 4 π geometry inside 20 polyethylene moderators lined with cadmium. 15 sets of 4 ^3He neutron detectors are connected in parallel, each set with a signal amplifier and a discriminator, to a signal mixer. The output of the signal mixer is fed via a low pass filter into a signal frequency analyzer. This frequency analyzer processes the signal pulse train for the two

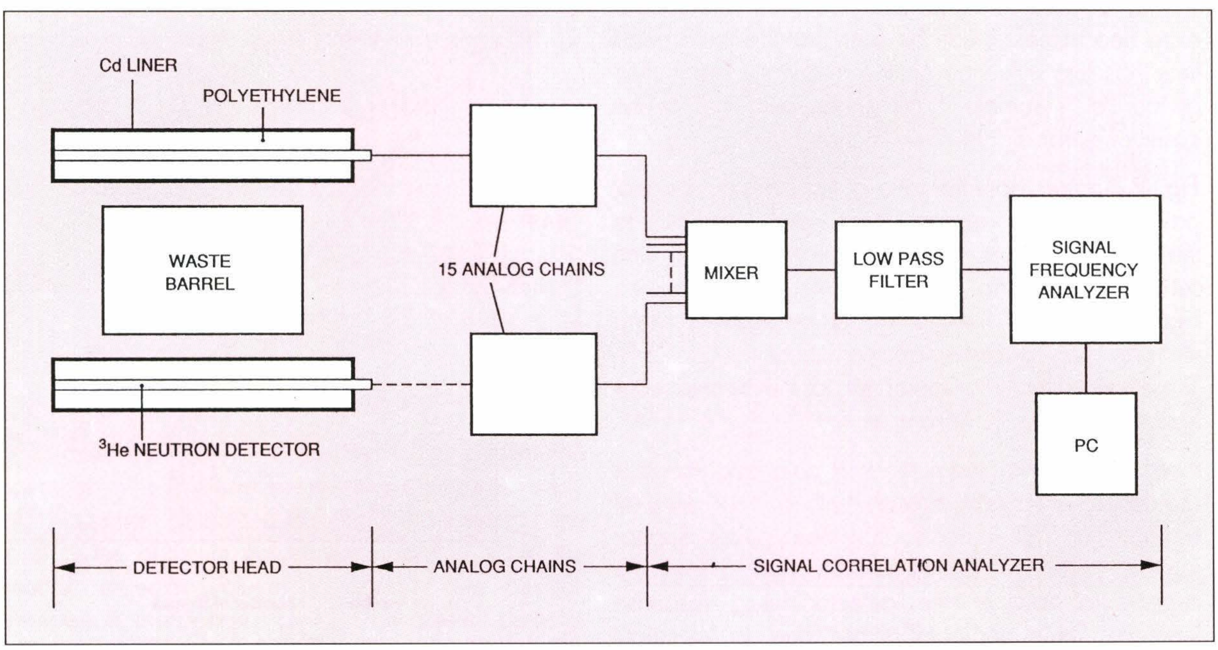

Fig. 9.1 Schematic view of the waste barrel monitor

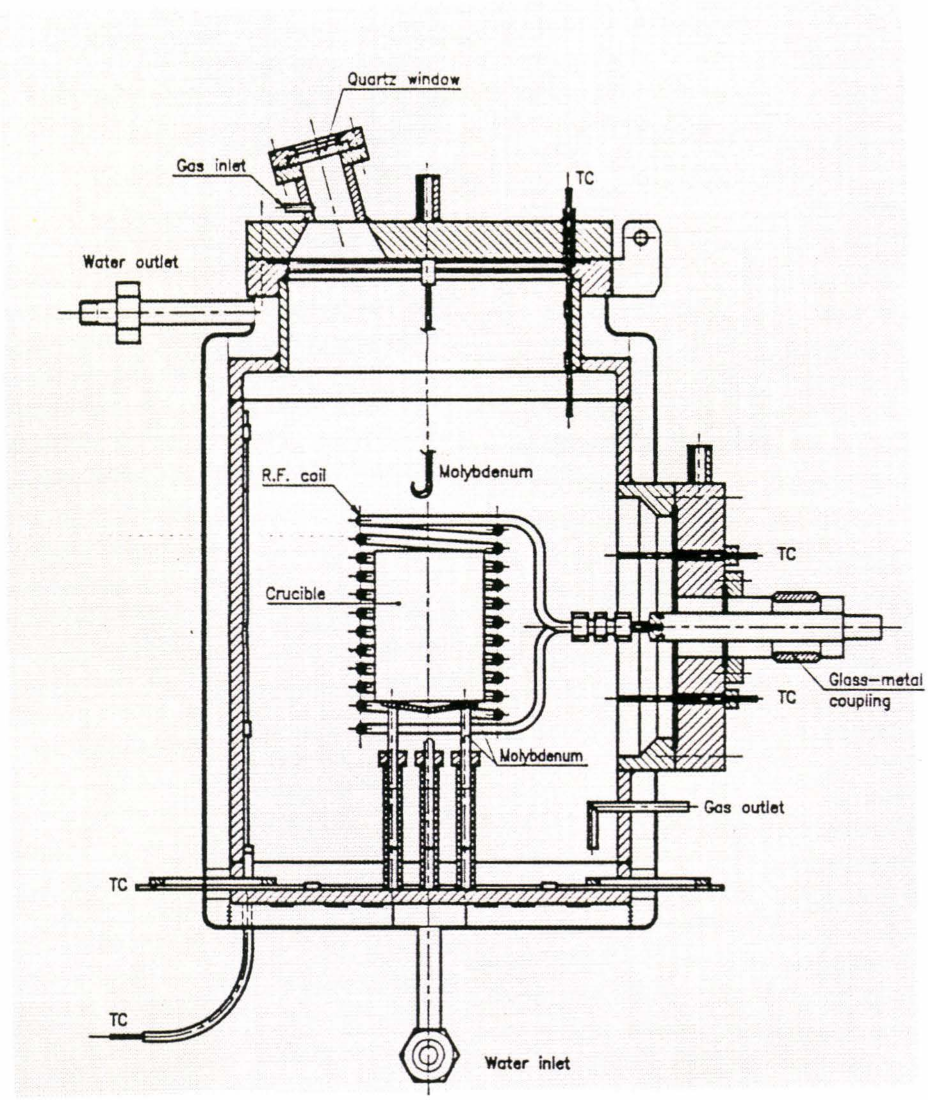

Fig. 9.9 Water cooled 304L induction furnace

4.6 PURIFICATION OF THE GASEOUS FEED STREAM TO FUEL CELL

The ideal gas feed to a Solid Polymeric Electrolyte Fuel Cell (SPEFC) [14,15] should be pure hydrogen. Only electrolytic hydrogen satisfies this condition. As a matter of fact, the usual gas feed is obtained by methane or methanol reforming. The composition of this reformed gas is approximately: 74% H_2, 24% CO_2 and 2% CO. In a large plant the CO_2 is removed by chemical absorption into alkaline solutions of ethanol amine. To avoid irreversible poisoning of the electrocatalysts due to the residual CO it is necessary to reduce the CO content of hydrogen to below 50 ppm [16].

The aim of the work is to study the possibility of achieving this purification level by means of a pressure swing adsorption (PSA) system [17] working at room temperature using a suitable zeolite as adsorber.

Zeolites are highly crystalline materials [18] with a regular array of silicon, aluminium and oxygen ions forming an anionic framework that encloses microporous channels of well defined sizes in the range of molecular dimensions. These channels are occupied by cations and water molecules, both having enough freedom of movement to permit reversible dehydration and cation exchange. Ion exchange is therefore, a property of most zeolites that enable a number of important applications. However, this property is also indirectly used for a tailored modification of the zeolite structure, hence properties, so that the resulting materials can be applied in catalytic or gas sorption processes. Indeed, the pore size of zeolites can be modified, in a controlled way, by cation-exchange. Exchangeable cations are normally located near the pore openings and this is particularly true for the monovalent cations. By exchanging the resident cation with another one, having a different diameter and/or electric charge, the pore openings may effectively change, modifying in such a way the molecular sieving [19]. Besides the molecular sieving effect, the adsorption and/or separation behaviour may also be controlled through the adsorption selectivity, that depends on the interaction between the adsorbate molecule and the exchangeable cation.

Progress up to the end of 1992

Three different type of zeolitic compounds have been selected and tested:

- Linde Type 4A: a low silica type zeolite (Si/Al = 1), having a three dimensional channel system composed by 4.1 Å windows connecting approximately spherical 11.4 Å cavities,
- Linde Type Y: a medium silica type zeolite (Si/Al≥2.5), having a three dimensional channel system composed by 7.4 Å windows connecting approximately spherical 11.8 Å cavities,
- Mordenite Type Na: a high silica type zeolite (Si/Al≥5), having a two dimensional channel system composed by straight 7.0x6.5 Å channels connected by short alternating channels (~3Å).

Products of two different manufacturers have been investigated, namely of:

- Philadelfia Quarz: NaMLP (PQ)
- Union Carbide: NaMLP (UOP)

The Na MLP (UOP) has been modified by partial exchange of Na with Ca to obtain the (Na, Ca)-Mordenite LP which has also been tested. Adsorption isotherms have been determined for CO (hydrogen is appreciably not adsorbed) at 238K and 300K. The results are reported in ***Fig. 9.10***. Chromatographic tests confirmed a high retention time for CO at room temperature especially if the (Ca,Na)-Mordenite used for the CO adsorption has been exchanged twice with Ca ions.

The Na-Mordenite Large Port has also been modified by exchange with Fe^{3+} ions to increase its adsorption selectivity towards CO. A detailed characterization of this product was planned for 1993, but unfortunately this exploratory research has been discontinued due to lack of funds.

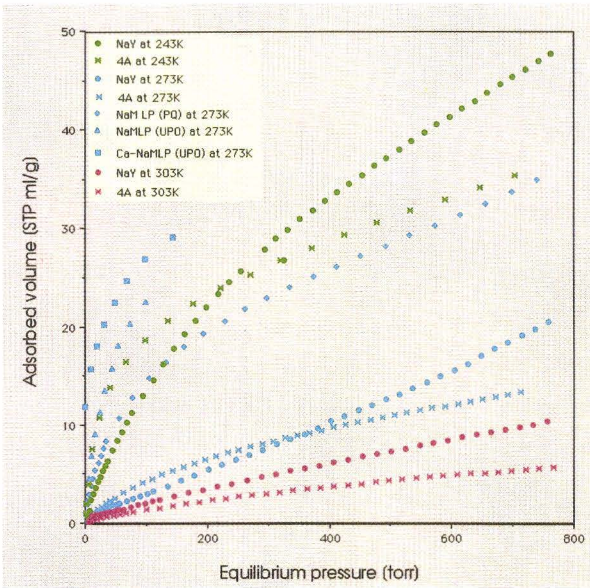

Fig. 9.10 Adsorption of carbon monoxide on Linde 4 A (low silica), Linde Na Y (medium silica), Na- and (Ca, Na)-Mordenite (LP) (high silica) substrates

Discussion of the preliminary results

The mordenite family shows the best adsorption characteristics. The difference between the two similar products coming from UOP and PQ is attributed to the higher crystallinity of the first one. The Ca-ion exchanged form has a larger adsorption capability at very low partial pressure. This means that a relatively large quantity of CO can be adsorbed to a low residual concentration. According to the increase of adsorption selectivity of polar molecules like CO with the increase of the charge of the exchangeable cation (as demonstrated by exchanging Na^+ with Ca^{2+}), it is believed that the product obtained by exchanging Na-Mordenite with Fe^{3+} ions should further increase the adsorption selectively for CO.

The results obtained so far, demonstrate that CO can be removed from H_2 by means of an engineered PSA system (possibly also a fixed bed) even at room temperature. The substrate should be of the Na-Mordenite family exchanged with bivalent or trivalent cations such as Ca^{2+} and Fe^{3+}. The acquisition of the physical parameters for designing and optimising the plant would require a supplementary investigation of approximately one year.

4.7 NEURAL NETWORKS FOR CHEMICAL REACTOR CONTROL

Batch chemical reactors are very complex systems from the point of view of application of control theory because they have pronounced non-linear dynamics, time-varying parameters and there is no steady state. Moreover, the operating conditions and type of chemical processes are changed constantly.

Therefore batch operation requires continuous corrections made by plant operators in order to maintain the control of the installation.

In the last years there has been extensive interest in the development of adaptive control systems able to compensate for variations in the characteristics of the process by adjusting their parameters automatically in such a way to avoid continuous tuning of the controllers. The application of neural networks (NN) technology to control batch chemical reactors seems very promising, because the adaptive nature of NN and their capability to learn from experience are basic characteristics of the "ideal" controller for these processes.

The work, started in 1991, has been pursued with emphasis on the use of more complex network architectures; in particular, two back-propagation NN connected in series were used to replace a traditional control system through simulation [20,21]. Even though FISIM (a numerical simulator of batch reactors) has been used to test the performances of the different NN architectures (*Fig. 9.11*), the possibilities to apply such an approach in a real process has been demonstrated. As a consequence, a heating/cooling loop in which neural controllers will be checked has been designed and is under construction.

In addition, the use of Neural Networks has been extended to another chemical engineering problem, i.e. the identification of complex kinetics [22] and heat transfer functions, and the early detection of loss of control of the reactor.

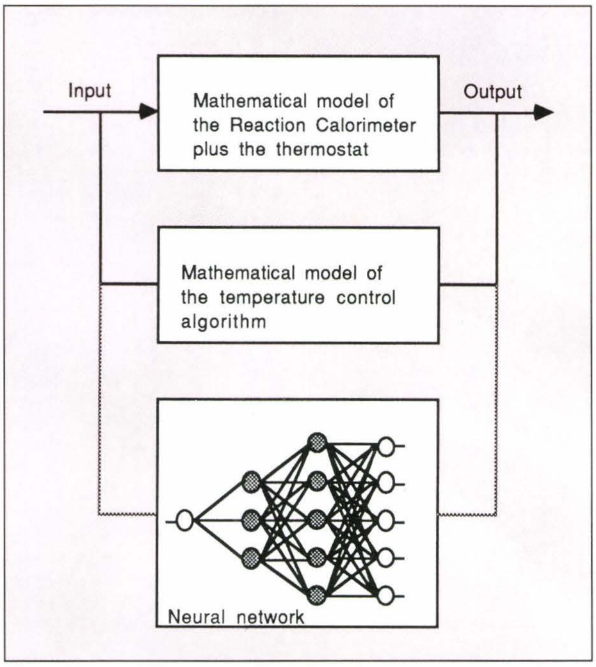

Fig. 9.11 Neural Network dynamic validation procedure

4.8 APPLICATION OF CHAOS THEORY TO BATCH AND SEMIBATCH CHEMICAL PROCESSES

The behaviour of a deterministic system for which prediction capacity is lost due to exponential sensibility versus initial conditions is called deterministic chaos. The techniques to study this behaviour are developed [23]. The main point of interest is the strange attractor, which is the part of the phase space (i.e. the space of the states which the system can assume) where the evolution is delimited when time goes to infinite. The strange attractor has ergodic properties and therefore the main types of ergodic systems are described.

The following techniques are used for the study of the attractor:

- Power spectrum, which can show the periodicity or chaotic behaviour of the system.
- Poincare section, which allows the reduction of the system variables to simplify its study [24].
- Dimensions (fractal, information, immersion, etc.), which give the geometric characteristics of the trajectory such as its density in the phase space [25].
- Lyapunov exponents and entropy, which give a measure of the divergence of the system's trajectory, quantify the chaotic behaviour and determine the rate of information losses.

The interdependencies of these techniques are pointed out. It was shown how to obtain the dimensions and the entropy from the Lyapunov exponents. These techniques have been applied to an oscillating chemical system, which can exhibit chaotic behaviour: the Belousov-Zhabothinsky reaction. A Lyapunov exponents algorithm has been adapted and implemented for this particular case [26]. These results are readily applicable for prediction of dangerous states of batch reactors [27]. The actual work for this activity will start in 1993.

STUDY OF THE TRUE STRESS-STRAIN RESPONSE OF PLAIN CONCRETE BY BUNDLE HOPKINSON'S BAR TECHNIQUE

The knowledge of the dynamic mechanical properties of plain concrete is of fundamental importance for the analysis of structural response under accident loading conditions, such as caused by impact, explosions or severe earthquakes. Experimental data on the true dynamic stress-strain response of plain concrete with real-size aggregate are rather scarse and this is due to the difficulty of capturing the phenomena of increasing damage up to fracture, especially under fast loading rate of the material.

The aim of the exploratory research is to investigate the potential of a new proposed experimental method, based on a bundle Hopkinson bar technique, for obtaining realistic data on the dynamic behaviour of plain concrete in tension and compression, including the softening range of the behaviour resulting from progressive cracking of the material.

Progress achieved at the end of 1992

The measurement of the true energy absorption capability of plain concrete will be performed with a bundle Hopkinson's bar whose concept is shown in *Fig. 9.12*. The real set-up consists of two sets of 25 aluminium bars to which a cubic concrete specimen with 20cm side will be glued. This test section will be installed in the Large Dynamic Test Facility (LDTF) as shown in *Fig. 9.13*.

Strain gauges applied to the individual bars in the bundle will provide local information on incident, transmitted and reflected pulses. The analysis of these signals is expected to yield information, instant by instant, on the true resisting cross section of the specimen during crack propagation, hence allowing a study of the true shape of the softening branch of the dynamic stress-strain diagrarnme and of the true energy absorption.

The achievements are the following:

- Design of the bundle Hopkinson's bar test section (March 1992) for tension testing
- Selection, purchase and setting-up of the dynamic instrumentation system for recording the strain gauges signals (October 1992)
- Construction of the bundle Hopkinson's bar test section for tension testing (December 1992).
- Patent proposal of the bundle Hopkinson's bar

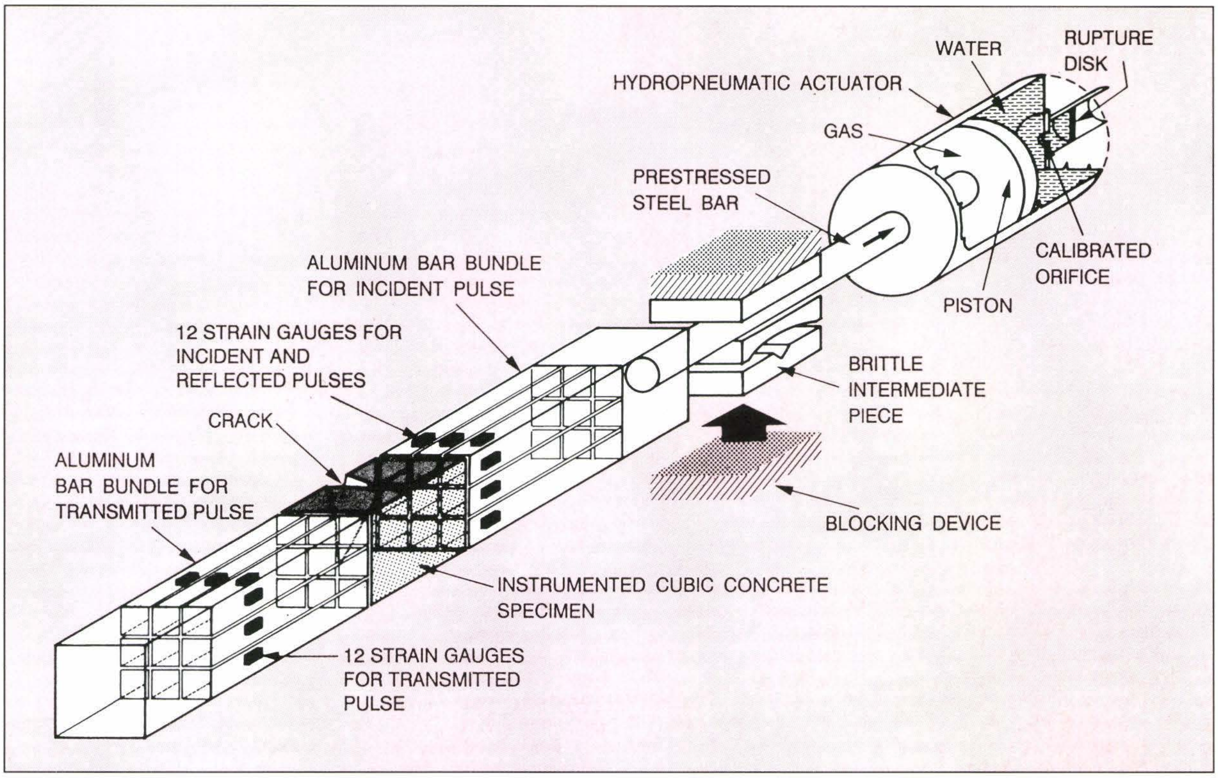

Fig. 9.12 Bundle Hopkinson bar for dynamic tension testing of plain concrete

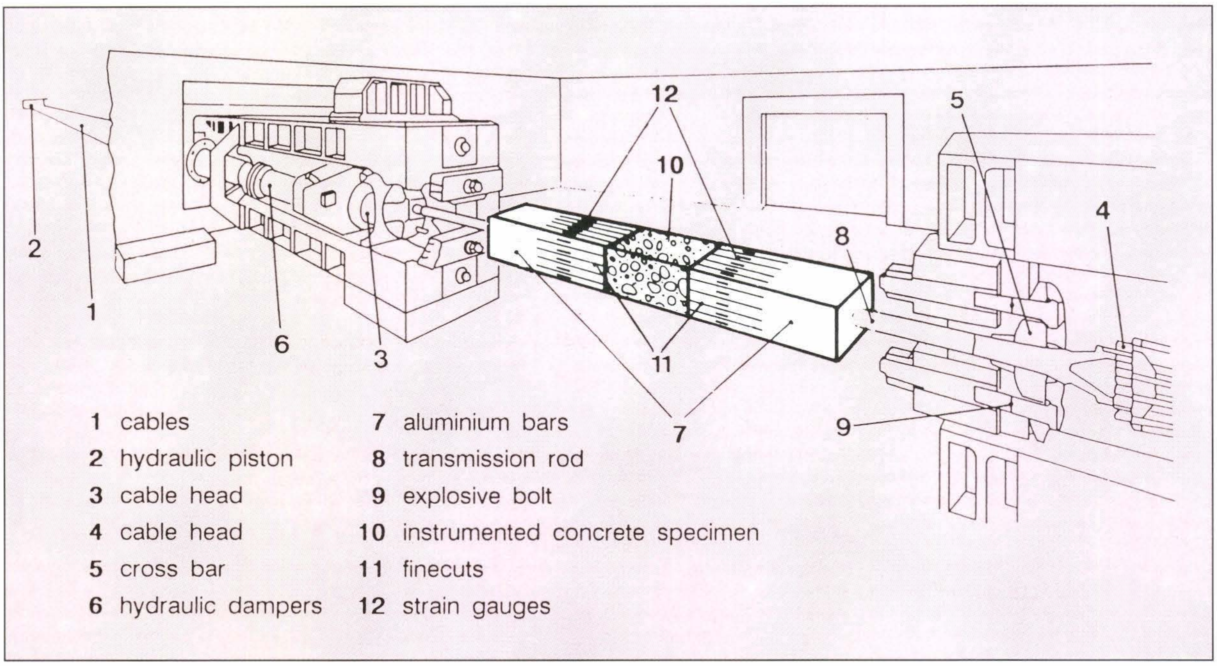

Fig. 9.13 Set-up of the bundle Hopkinson bar in the large dynamic testing facility

test section registered at the Luxembourg patent office under the number P/2352 entitled "Universal test apparatus" (May 1992).

- Design of the Hopkinson's bar test section for compression testing (December 1992).
- Development of a measurement programme on plain concrete in collaboration with ENEL-CRIS, ENEA-DISP, Nuclear Electric, CRIEPI, Politecnico di Milano, University of Karlsruhe which will participate to the programme with financial and scientific support, during the years 1993 and 94.

References

[1] RIEF H., TAKAHASHI H. - A critical Review of Accelerator-based Transmutation Systems - invited paper at the OECD-Nuclear Energy Specialists' Meeting on Accelerator-based Transmutation, Wuerenlingen (CH), March 24-26, 1992

[2] TAKAHASHI H., RIEF H. - The Energy Requirement for Transmuting Fission Products - OECD/NEA Second General Meeting of the International Information Exchange Programme on Actinide and Fission Product Separation and Transmutation, ANL, Nov. 11-13, 1992

[3] TAKAHASHI H., RIEF H. - The Energy Requirement for Transmuting and Isolation Fission Products - 1992 ANS/ENS International Meeting, Chicago, Nov. 15-20, 1992

[4] RICCHENA R. - Calcoli per l'installazione BNCT di Petten con metodo deterministico - Pavia-Workshop on BNCT, 13th June 1991

[5] RIEF H., VAN HEUSDEN R., PERLINI G. - Generating Epithermal Neutron Beams for BNCT in Small Reactors - 5th International Symposium on Neutron Capture Therapy for Cancer, Columbus (Ohio), Sept. 12-19, 1992

[6] RICCHENA R. - Research on Optimal Arrangement of the Neutron and Gamma Filter in the HB 11 Channel for Boron Neutron Capture Therapy - Technical Note, October 1992

[7] HAGE W., CIFARELLI D.M. - On the factorial moments of the neutron multiplicity distribution of fission cascades - Nucl. Instr. and Meth. A236, pp. 165-177, 1985

[8] HAGE W., CIFARELLI D.M. - Correlation analysis and neutron count distributions registered updating dead time counter for the assay of fissile materials - Nucl. Sci and Eng. 112, pp. 136-158, 1992

[9] PEDERSEN B., Ph.D. Thesis - Applications of the Passive Neutron Correlation Technique to the Assay of Pu in Radioactive Waste Reactor Centre - Imperial College of Science, Technology & Medicine, University of London (in preparation)

[10] MOUKASSI M., STEINMETZ P., DUPRE B., GLEITZER C. - Study of the Slowing Down at Moderate Temperature, in the Reduction with Hydrogen of Iron Oxide and Ores - Proc. of the World Hydrogen Energy Conference IV, Pasadena, California, USA, 13-17 June, 1982, Vol. 2, Pergamon Press (1982)

[11] OLETTE M., BESSIER J., RIST A., GLEITZER C., FAUCHAIS P. - L'Hydrogène en Siderurgie - Entropie N° 116/117 (1984)

[12] DAVIS C.G., McFARLIN J.F., PRATT H.R. - Direct Reduction Technology and Economics - Ironmaking & Steelmaking, Vol. 9 N° 3 (2) 93

[13] MULLER R. - The Use of Hydrogen Plasma Processes in Petrochemical and Iron-smelting Industries - see ref. [10]

[14] KORDESH K. - Fuel Cells: The Present State of Consideration of the Alkaline Hydrogen/Oxigen Systems - Hydrogen Energy Progress, 6th World Conference, Vienna, Austria, Vol. 3, pp. 1221-1228, 20-24 July, 1986

[15] KORDESH K.. - The Choice of Low-Temperature Hydrogen Fuel Cells: Acidic or Alkaline? - Hydrogen Energy Progress, 4th World Conference, Pasadena, California, USA, pp. 1139-1148,13-17 June, 1982

[16] CASATI D. - Problematiche tecniche ed economiche relative al trattamento del combustibile negli impianti a fuel cells - Atti del convegno "Celle a combustibile: stato dell'arte", Milano, Italia, 12 dicembre 1989

[17] RUTHVEN D.M. - Principles of Absorption and Absorption Processes - J. Wiley & Sons, pp. 361-379, 1984

[18] BRECK D.W. - Zeolite Molecular Sieves: Structure, Chemistry and Use - J. Wiley & Sons, pp. 29-132, 1974

[19] TOWNSEND R.P. - Ion Exchange in Zeolites: Some Recent Developments in Theory and Practice - Pure & Appl. Chem., Vol. 58, n° 10, pp. 1359-1366, 1986

[20] ZALDIVAR J.M., PANETSOS F., HERNANDEZ H. - Control of batch reactors using neural networks - Chemical Engineering and Processing, 31, pp. 173-180, 1992

[21] PANETSOS F., ZALDIVAR J.M. - Use of feed-forward neural networks in batch reactor control - (submitted to IEEE Transactions on Neural Networks)

[22] ZALDIVAR J.M., HERNANDEZ H., MOLGA E., GALVAN I.M., PANETSOS F. - The use of neural networks for the identification of kinetic functions of complex reactions - accepted by the Third European Symposium on Computer Aided Process Engineering, Graz, 5-7 July, 1993

[23] STROZZI F., HERNANDEZ H., ZALDIVAR J.M. - Overview of the methods to analyse dissipative chaotic systems with applications to chemical processes - Technical Note to be published, 1993

[24] WIGGINS S. - Introduction to applied nonlinear dynamical systems and chaos - Springer-Verlag, New York, 1990

[25] MAYER-KRESS G. - Dimensions and entropies in chaotic systems - Springer-Verlag, Berlin, 1986

[26] WOLF A., SWIFT J.B., SWINNEY H.L., VASTANO J.A. - Determining Lyapunov exponents from a time series - Physica 16D, pp. 285-317, 1985

[27] GIONA M. - Functional reconstruction of oscillating reaction: prediction and control of chaotic kinetics - Chem. Eng. Science, Vol. 47, No. 9-11, pp. 2469-2474, 1992

5 HUMAN RESOURCES

5.1 Structure and distribution

5.1 STRUCTURE AND DISTRIBUTION

Table 10.1 Institute structure

SAFETY TECHNOLOGY INSTITUTE

Director: H.F. Holtbecker

Deputy Directors: P. Fasoli Stella, H. Dworschak

Senior Scientists:
M. Cambini
F. Mannone
G. Pierini

Coordination & Secretariat:
Secretariat: M.E. Marengo, B. Del Vitto
QA/QC: L. Ferrario

Thermodyn. & Radiat. Physics Unit P. Fasoli-Stella	Process Engineering Unit C. Busse	Applied Mechanics Unit J. Donea	Nuclear Fuel Cycle Unit H. Dworschak	Nuclear Experiments Unit G. Barbera, F. Tinagli*	In-pile Tests Unit (at Cadarache) P. von der Hardt	Nucl. Mat. Condit. & Transport Unit S. Zanella
Severe Accident Studies A. Markovina	Computational Proc. Analysis H. Städtke	Earthquake Engineering P. Jones	Nuclear Measur. & Actinide Monit. L. Dierckx	ETHEL Laboratory G. Vassallo (Laboratory Head)		Radioact. Waste Conditioning Sector R. Gritti
Radiation Physics H. Rief	Ind. & Environm. Hazards Analysis R. Nijsing	Computational Mechanics J.P. Halleux	Integrated Safeguards Sector S. Guardini	PERLA Laboratory S. Guardini (NFCU)		
Fluid Dynamics Modelling A. Benuzzi	Experimental Fluid Dynamics G. Krüger	Experimental Mechanics G. Verzeletti	Nuclear Chemical Engineering Sector B. Hunt	PETRA Laboratory S. Bertelli (Laboratory Head)		
Thermomech. Tests H. Hohmann	Material Properties C. Bassani		Fusion Materials Studies Sector F. Reiter	Hot cells Sector J. Cauwe		
Fission Product Transport A. Jones	Techn. Support L. Fritz		Fluids Separation & Struct. Analysis Sector F. Toci	Plant Maintenance Sector P. Bastianini		
Laboratory for Aerosol Studies G. De Santi			Plant Design & Modelling Sector J. Bourdon	Computational Analysis Sector A. Caruso		
Therm. Hydraulics C. Addabbo						

* Responsible for Licensing & Operations

Table 10.2 Institute staff (status 31.12.1992)

	Scient./ Techn. Staff	Admin. Staff	Authorized Recruitm. 1992		People who left in 1992		Grantholders		Visiting Scientists		Experts Seconded		Auxil. Agents
			ST	Adm.	ST	Adm.	present	addit. expected arrivals	present	addit. expected arrivals	present	addit. expected arrivals	
Direction	6	2							–				
Thermodynamics and Radiation Physics	60	5	1	–	–	–	2	–		–	1	–	1
Process Engineering	49	4	1	–	2		5	–	1		2	–	–
Applied Mechanics	27	2	–	–	1	–	2	–			–	–	–
Nuclear Fuel Cycle	53	4	1	–	1	–	3				1		–
Nuclear Experiments	51	2	–	–	–	–	–	–	–	–	–	–	–
In Pile Tests	3	1	–	–	–	–	–	–	–	–	–	–	–
Nucl. Mat. Condit. and Transport Unit	28	10	–	–	–	–	–						–
	277	30	3	–	4		12		1		4	–	1

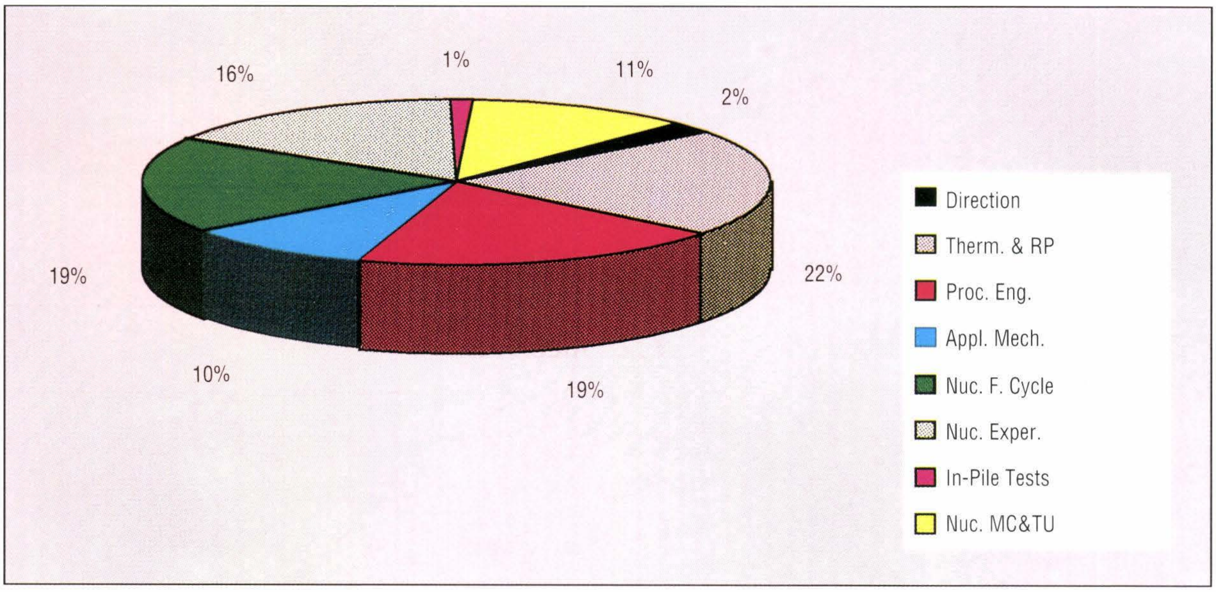

Fig. 10.1 Institute staff (status 31.12.1992), distribution (%)

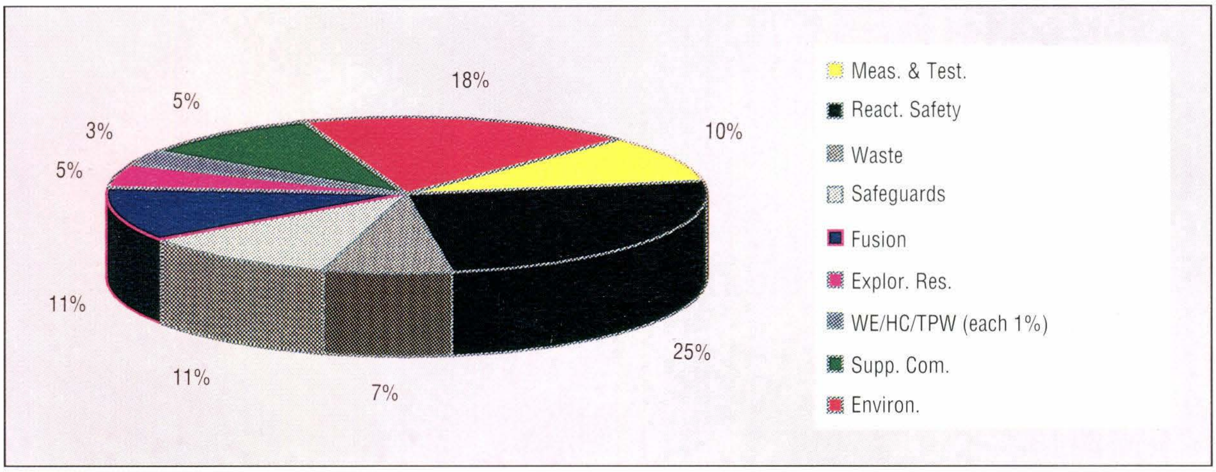

Fig. 10.2 STI - financial resources (1992), distribution (%)

6 PUBLICATIONS AND EVENTS

LIST OF PUBLICATIONS

Specific Programmes

Reactor Safety

CONTRIBUTIONS TO PERIODICALS AND MONOGRAPHS

MAGALLON D., HOHMANN H., SCHINS H. - Pouring of 100 kg-Scale Molten UO_2 inot Sodium - Nuclear Technology, Vol. 98 (1992), pp. 79-90, ART 29774

PROCK J., OHLMER E., LABEIT M. - On-Line Test of Signal Validation Software of the LOBI-MOD2 Facility in Ispra, Italy-Nuclear Technology, Vol. 97 (1992), pp. 52-62, ART 40246

TECHNICAL EUR REPORTS

SHEPHERD I. et al. - PHEBUS-FPTO: calculations for the Reference Scenario - Volume 1: The Bundle, EUR 14225EN, 1992

BENSON C. - Some Chemistry Related Aspects of the Phebus-FP Tests - EUR 14328/EN, 1992

BENSON C. - Iodine Chemistry in the Reactor Containment Building during an LWR Severe Accident -Note to the First Meeting of the European Severe Accident Chemistry Group, held at Ispra on 22-23 May 1991, EUR 14341/EN, 1992

BENSON C. - High-Temperature Chemistry - Note of the Second Meeting of the European Severe Accident Chemistry Group held in Grenoble on 28-29November 1991 - EUR 14489/EN, 1992

BENSON C. - The Influence of Organic Paints on Containment Iodine Behaviour under Severe Reactor Accident Conditions - EUR 14565/EN, 1992

ADDABBO C. - Overview on the LOBI Project - Proceedings of the LOBI Seminar, Arona, Italy, EUR 14174 EN, April 1992

ANNUNZIATO A. - Synopsis of LOBI-MOD2 Experimental Results - Proceedings of the LOBI Seminar, Arona, Italy, EUR 14174 EN, April 1992

ADDABBO C. (Editor) - Proceedings of the LOBI Seminar - Arona, Italy, EUR 14174 EN, April 1992

CONTRIBUTIONS TO CONFERENCES

SHEPHERD I.M., SERRE F. - Precalculations for the Bundle for the First Phebus-FP test FPT-0 - IAEA Technical Committee Meeting, Behaviour of Core Materials and Fission Product Release in Accident Conditioning Light Water Reactors - Aix-en-Provence, 16-19 March 1992

JONES A.V., SHEPHERD I. - ESTER - a new approach in source term code development - Deutsche Atomforum Jahrestagung Kerntechnik '92 (Inforum GmbH, Bonn), May 1992

SHEPHERD I., JONES A., HERRANZ L., ALCAMI M. - Scoping Calculations for the Containment in Phebus FPT-1 - XVIII Reunion Anual, Spanish Nuclear Society, Jerez, October 1992

HERRANZ L., VICENTE M-C., JONES A.V., SHEPHERD I. - Contributions to the Pretest Analysis of the Phebus-FP Tests - XVIII Reunion Anual, Spanish Nuclear Society, Jerez, October 1992

JONES A. - Multiphase Flow Modelling and Applications at the CEC Joint Research Centre - Proceedings of the European Computational Fluid Dynamics Conference, 7-11 September 1992, Bruxelles (B), Comutational Methods in Applied Siences Ch., Eslevier Science Publ. (1992), pp. 191-200, Hirsch (Ed.), ORA/PRO 36888

JONES A.V., SHEPHERD I. - ESTER, a European Severe Accident Code System - USNRC 20th Water Reactor Safety Information Meeting, Bethesda (MD, USA), October 1992

MAILLIAT A., SERRE F., JONES A.V., SHEPHERD I. - The Calculation Programme to Prepare the First Phebus-FP Tests - USNRC 20th Water Reactor Safety Information Meeting, Bethesda (MD, USA), October 1992

ADROGUER B., MAILLIAT A., JONES A.V., SHEPHERD I. - Calculation of the Phebus-FP experiments - 3rd Workshop on Severe Accident Research in Japan, Tokyo, November 1992

ZEYEN R. - Instrumentation for Integral Severe Accident Simulating Experiments - In:Proceedings of the Specialists Meeting on Instrumentation to Manage Severe Accidents, OECD, CSNI, ORA/PRO 36583, Köln (D), 16-18 March 1992

SCHEURER H., ZEYEN R., KTORZA C., CLEMENT B. - Integral Source Term Experiment PHEBUS FP, with Special Emphasis to In-Pile and Circuit Instrumentation - In: Proceedings of the Intern. Conference on Irradiation Technology, CEA, CEC-JRC Petten, ORA/PRO 36612, Saclay (F), 20-22 May 1992

RIEF H., VAN HEUSDEN R., PERLINI G. - Generating Epithermal Neutron Beams for BNCT at TRIGA Reactors - In: Proceedings of the Fifth Intern. Symposium on Neutron Capture Therapy for Cancer, ISNC, ORA/PRO 36823, Columbus, Ohio (USA), 13-17 September 1992

ZEYEN R., WILHELM H., LUCAS M. - The PHEBUS FP Integral Souce Term Experimental Project, with Emphasis on Iodine Selective Filtering - In: Proceedings of the 22nd Nuclear Air Cleaning and Treatment Conference, DOE/NRC, ORA/PRO 36958, Denver, Col. (USA), August 1992

CALVO M., GOMEZ F., KRISCHER W., MARCOS M., RAMSDALE S. - The EC-Commission's Contribution to the Research on FP Pool Scrubbing under Severe Accident Conditions - XVIII Reunión Anual, SNE, Spanish Nucl. Soc., Jerez de la Frontera (E), Resúmenes de las Ponenzias, D.L. M-34137, pp. 221-225, Macula-Madrid, ORA/PRO 37116, 28-30 October 1992

MEYER-HEINE A., FASOLI P. - Experimental Programmes - Training Seminar on Sevee Accidents and Accident Management in LWRs, OECD & CEA, ORA 36706, Lyon (F), 23-27 March 1992

KRISCHER W. - Status of Work of the EC Research Activity on Pool Scrubbing - 9th European Pool Scrubbing Group Meeting, ORA 37176, ENEL, Milano (I), 7 October 1992

VON DER HARDT P., GUILLEMARD B. - The PHEBUS Fission Product Progream - In: Procedings of the Third Workshop on Severe Accident Research in Japan, ORA/PRO 37185, JAERI-NUPEC, Tokyo (Japan), 4-6 November 1992

CAPITAO J., DROSSINOS Y.- Aerosol Interactions and Transport during the First PHEBUS Experiment According to the VICTORIA and RAFT Computer Codes - European Aerosol Conference, 6-11 September 1992, Oxford (UK), Journal of Aerosol Science, Special Issue, ORA/ART 37155

BUSSE C. - Heat Pipe Science - 8th Intern. Heat Pipe Conference, Chinese Acad. Sci, 14-18 September 1992, Beijing (PR China), ORA 36887

ZEYEN R., HUEBER C. - Iodine Calculations and Measurements in the PHEBUS FP Containment - 4th Meeting of the European Severe Accident Chemistry Group, ORA 37288, JRC/ENEL, Milano (I), 12 December 1992

BENUZZI A. - Controllo delle Sovrappressioni; Parte 1A: Metodi per il Dimensionamento dei Sistemi di Sfiato - In: Atti del Corso: Sicurezza dei Reattori Chimici Batch e dei Serbatoi di Stoccaggio, ORA/PRO 36891, IBM/MATEC, Milano (I), 7-8 maggio 1992

BENUZZI A. - Metodi per il Dimensionamento dei Sistemi di Sfiato - Chimica e Sicurezza: Contributo dei Metodi Computerizzati, ORA 36934, SChI, San Donato Milanese, 10 giugno 1992

EUSEBI M., PAROZZI F., VALISI M., CAPITAO J.A., DE SANTI G.F. - Preparatory Calculations for a New Experimental Program on Dry Aerosol Resuspension Mechanisms (STORM Project) - Paper presented at the European Aerosol Conference, Oxford (UK), September 1992

ANNUNZIATO A., ADDABBO C., BRIDAY G, DERUAZ R., JUHEL D., KUMAMARU H., KUKITA Y., MEDICH C., RIGAMONTI M. - Small Break LOCA Counterpart Test in the LSTF, BETHSY, LOBI and SPES Test Facilities - Proceedings of the NURETH5 Conference, Salt Lake City, USA, September 1992

D'AURIA F., GALASSI G., ANNUNZIATO A., ADDABBO C. Comparative Analysis of Accident Management Procedures in the LOBI and SPES Test Facilities - Proceedings of the NURETH5 Conference, Salt Lake City, USA, September 1992

WORTH B., PELLISSIER M., FRANCHELLO G. - Recent experience with CATHARE2 calculations at JRC-Ispra;' Presented at the LOBI Seminar, Arona, March 31-April 2, 1992

STÄDTKE H. - International Code Assessment Activities in the Context of the LOBI Project - Proceedings of the LOBI Seminar, Arona, March 31-April 2, 1992

DE SANTI G., BACHLER J. - Operational Experience in the LOBI-MOD2 Test Facility - Proceedings of the LOBI Seminar, Arona, March 31-April 2, 1992

WORTH B. - Synposis of Code Calculations and Assessment Activities within the LOBI Project - Proceedings of the LOBI Seminar, Arona, March 31-April 2, 1992

CORRADINI M., HOHMANN H. - Multiphase Flow Aspects of Fuel-Coolant Interactions in Reactor Safety Research - Specialists' Meeting on Transient Two-Phase Flow "Current Issues in System Thermal Hydraulics", OECD, 6-8 April 1992, Aix-en-Provence (F), ORA 36722

SCHINS H., HOHMANN H., BUERGER M., VON BERG E., CHO S. - Breakup of Melt Jets in a Water Pool as a Key Porcess for Analysis of Lower RPV-Head Failure during Core Melt Accidents in LWR - Proceedings of the Jahrestagung Kerntechnik '92, Deutsches Atomforum, 5-7 May 1992, Karlsruhe (D), Tagungsbericht (1992), pp. 159-162, ORA/PRO 36654

COMMUNICATIONS*

ANNUNZIATO A. - Quick Look Report on LOBI-MOD2 Test BT-17 - CEC-JRC, Ispra, COM 4341, July 1992

ADDABBO C., LEVA G., ANNUNZIATO A. - Experimental Data Report on LOBI-MOD2 Test BL-34 - CEC-JRC, Ispra, COM 4335, July 1992

ADDABBO C., LEVA G. - Quick Look Report on LOBI-MOD2 Test BL-16 - CEC-JRC, Ispra, COM 4342, October 1992

ADDABBO C., LEVA G., ANNUNZIATO A. - Experimental Data Report on LOBI-MOD2 Test BL-44 - CEC-JRC, Ispra, COM 4336, November 1992

ADDABBO C., LEVA G., ANNUNZIATO A. - Experimental Data Report on LOBI-MOD2 Test BT-17 - CEC-JRC, Ispra, COM 4344, November 1992

ADDABBO C., LEVA G., ANNUNZIATO A. - Experimental Data Report on LOBI-MOD2 Test BL-40 - CEC-JRC, Ispra, COM 4345, December 1992

ANNUNZIATO A. - Quick Look Report on LOBI-MOD2 Test BL-44 - CEC-JRC, Ispra, COM 4343, December 1992

TECHNICAL NOTES*

SHEPHERD I.M., RIEGER T., VAN RIJ H.M. - The Response of the Phebus-FP Containment Vessel to Thermalhydraulic Boundary Conditions - Technical Note No. I.92.18, January 1992

SANDERS J. - Minutes of the 5th Meeting of the Safety Technology Institute Advisory Board, Ispra, 10-11 October 1991 - Technical Note No. I.92.03

SANDERS J. - Minutes of the 6th Meeting of the Safety Technology Institute Advisory Board, Ispra, 25-26 May 1992 - Technical Note No. I.92.59

* Restricted distribution

YERKESS A. - Scoping Structural Calculations of the PHEBUS Test Section Submitted to Transient Loadings - Technical Note No. I.92.11

YERKESS A. - Structural Response of the TERMOS Vessel to Dynamic Loadings - Technical Note No. I.92.25

BENUZZI A. - The JRC Research Project in the Area of Indoor Ventilation and Pollutant Transport Modelling - Technical Note No. I.92.95

MAGALLON D. et al. - Scoping Test Data Report - JRC-Ispra, Technical Note No I.92.135, December 1992

WIDER H. et al. - Quick Look Report on the Scoping Test - JRC-Ispra, Technical Note No I.92.139, December 1992

ANNUNZIATO A. - Analysis of the FARO facility initial pressurization behaviour during the 150 kg test - Technical Note, 09.12.1992

HOHMANN H. et al. - KROTOS 26 to KROTOS 30: Experimental Data Collection - JRC-Ispra, Technical Note No. I.92.115, November 1992

ANNUNZIATO A. - Effectiveness of Secondary Feed and Bleed in the LOBI-MOD2 Test Facility -CEC-JRC, Ispra, Technical Note No. I.92.19, March 1992

PELLISSIER M., WORTH B. - CATHARE2 Base Input Data for LOBI/MOD2 - JRC-Ispra, Technical Note No. I.92.17, February, 1992

PELLISSIER M., WORTH B. - CATHARE2 Calculation of LOBI Test A2-81 - JRC-Ispra, Technical Note No. I.92.24, March, 1992

PELLISSIER M., WORTH B. - CATHARE2 Calculation of LOBI Test BL-12 - JRC-Ispra, Technical Note No. I.92.44, May, 1992

PELLISSIER M., WORTH B. - CATHARE2 Calculation of LOBI Test BL-34 - JRC-Ispra, Technical Note No. I.92.64, June, 1992

PELLISSIER M., FRANCHELLO G., WORTH B. - CATHARE2 Calculation of LOBI Test BT-01 - JRC-Ispra, Technical Note No. I.92.109, November, 1992

PELLISSIER M., FRANCHELLO G., WORTH B. - CATHARE2 Calculation of LOBI Test BT-12 - JRC-Ispra, Technical Note No. I.92.110, November, 1992

RUEL F. - Numerical Simulation of Reacting Gas Flows: Assessment Calculaions of Transonic Multidimensional Flows - Technical Note No. I.92.01

RUEL F. - Partial Equilibrium Solvers for Reactive Transonic Flows: Basic Algorithms - Technical Note No. I.92.125

MAGALLON D. - FARO LWR Programme - Scoping Test Data Report - Technical Note No. I.92.135

ANNUNZIATO A. - Break Mass Flow Rate Evaluation in the Simulation of a SGTR Accident - Technical Note No. I.92.02

WIDER H. - The FARO/LWR Experimental Programme - Quick Look Report on the Scoping Test - Technical Note No. I.92.139

HULD T. - Numerical Simulaiton of Reacting Gas Flows: Assessment Calculations of Deflagrations/Detonations - Technical Note No. I.92.66

AREIRA-CAPITAO J. - RAFT Sensitivity Studies for PHEBUS-FPT0 - JRC Contribution, Technical Note No. I.92.82

SPECIAL PUBLICATIONS

ADDABBO C. -LOBI: LWR Off-Normal Behaviour Investigations - CEC-JRC, Ispra, S.P. I. 92.14, March 1992

HOLTBECKER H., (Ed.) - Safety Technology Institute - S.P./I.92.28

ADDABBO C. - LOBI Project - Publications (Status March 1992) - S.P./I.92.08

WORTH B. - Energy Issues and Nuclear Energy - S.P./I.92.01

PETRUCCIOLI H. - LOBI Project: Documentation and Publications CEC-JRC, Ispra, S.P. I.92.08, March 1992

Safeguards and Waste

CONTRIBUTIONS TO PERIODICALS AND MONOGRAPHS

HAGE W., CIFARELLI D.M. - Correlation analysis with neutron count distribution for a paralyzing dead-time counter for the assay of spontaneous fissioning material - Nucl. Sci. and Eng. 112, pp. 136-158, 1992

SCHILLEBEECKX P., WAGEMANS C., DERUYTTER A., BARTHELEMY R. - Comparative study of the fragments' mass and energy characteristics in the spontaneous fission of 238 Pu and 242 Pu and in the thermal-neutron-Nuclear Physics - A545, p. 623, 1992

TECHNICAL EUR REPORTS

NAPIER S., SCHILLEBEECKX P. - PHONID 3b measurements of uranium waste - EUR 14551 EN, 1992

Proceedings of the International Workshop on Calorimetry - JRC, PERLA Laboratory, Ispra (I), EUR 14307/EN, March 1992

GUARDINI S. et al. - Assessment of Plutonium Calorimetry Performances at PERLA - Proceedings of the International Workshop on Calorimetry - JRC, PERLA Laboratory, Ispra (I) EUR 14307/EN, March 1992

VOCINO V., BINDA F.,CARAVATI G. , D'ADAMO D., FARESE N., MAUCQ T., REMORINI B. - Combined use or neutron correlation technique, calorimetry and gamma spectroscopy for the non destructive mass determination of the Plutonium isotopes in nuclear fuel - Proceedings of the International Workshop on Calorimetry - JRC Ispra, EUR 14307 EN, March, 1992

VOCINO V., MAUCQ T., FARESE N., VERRECCHIA G., NEBULONI M. - PECC: User's Manual, EUR 14564/EN (1992)

ALBERMAN A., DIERCKX R., NOLTHENIUS H., VOORBRAAK W. - Status of Radiaiton Damage Dosimetry for Fusion Materials Testing in Reactors - EUR 14630/EN (1992)

GUARDINI S. - PERLA Safeguards Performance Laboratory - EUR 14783/EN (1992)

CONTRIBUTIONS TO CONFERENCES

BONDAR L., D'ADAMO D., DIERCKX R., HAGE W., HUNT B., PEDERSEN B.,. SCHILLEBEECKX P, VOCINO V. - Measurement facilities for radioactive waste at the JRC Ispra establishment - 14th Annual ESARDA meeting, Salamanca, Spain, 5/8 May, 1992

BONDAR L., D'ADAMO D., DIERCKX R., HAGE W., HUNT B., PEDERSEN B., SCHILLEBEECKX P., VOCINO V. - Measurement facilities for radioactive waste at the JRC Ispra establishment - 14th Annual ESARDA meeting, Salamanca, Spain, 5/8, May, 1992

ZUKOSKY P.A., HODGES J.K., HUCKINS R.J., MERCIER M.T., ZALOKAR G.S., KOSKELO M.J., SMITH B.G.R., SWINHOE M., FRANKLIN M., BUTEZ M. , MARTIN-DEIDIER L., RICHARD J.P., TALLEC M., PETIT S. - A Host Computer System for Distributed, Unattended Plutonium Safeguards - Proc. of the 33rd Annual Meeting of the Institute of Nuclear Materials Management, Orlando, USA, 19-22 July, 1992

HUNT B., THORNTON M., SUDA S., KEISCH B. - Demonstration of the Pressure-Volume Authenticator on the PETRA Input Accountancy Tank -33rd Annual Meeting of the Insitute of Nuclear Materials Management, INMM, 19-22 July 1992, Orlando, Fla. (USA), ORA 36765

DIERCKX L. - Neutron Spectrum Adjustment Based on Activation Techniques, Methods Sensitivity and Uncertainties - Interregional Training Course on Applications of Nuclear Data and Measurement Techniques in Nuclear Reactor and Personal, 15 June - 10 July 1992, Obninsk (CIS), ORA 36789

GUARDINI S. - Results and Conclusions from the ESARDA Technical Workshop on NDA Techniques Applicable to Safeguarding Nuclear Materials in Waste - Technical Programme Committee Meeting, INMM, 19-22 July 1992, Orlando, Fla. (USA), ORA 36882

TECHNICAL NOTES*

BONDAR L., HAGE W., PEDERSEN B., SWINHOE M., HAAS E., LEDEBRINK F.W.,. AMELING W, KLEINEKORTE K., KARAVAS A. - Pu waste measurement in a mox fuel fabrication plant by neutron correlation methods - Technical Note N° 1.92.99, 1992

BARDELLI R., BECKER L., LEZZOLI L., SCHILLEBEECKX P., WENG U. - Measuring Characteristics in Phonid 3b - Technical Note No. 1.92.54, 1992

EDWARDS R.A.H., BOTTELIER P., FACELI R., ELEN J.D. - FRUST Project Test Report: Tensile Testing of Irradiated Samples from Welds in Solution Annealed 316L Austenitic Steel Plate, Technical Note No. I.92.114

SPECIAL PUBLICATIONS

GUARDINI S. - PERLA, Instruments and Systems - S.P./I.92.25

Fusion technology and safety

CONTRIBUTIONS TO PERIODICALS AND MONOGRAPHS

RENDA V., SORIA A., PAPA L. - Thermal Transients Due to Plasma Sweeping on the Monoblock Divertor Plate for ITER - Fusion Technology, Vol. 22 (1992), pp. 490-500, ART 40237

REITER F., TOMINETTI S., PERUJO A. - Tritium Transport in the Water Cooled Pb-17Li Blanket Concept of DEMO - Fus. Eng. Desig. 15, pp. 223-234, 1992

GORAN-LOVESTAM N.E., SWIETLICK E., WATJEN U., LOUWERIX E., PERUJO A., RIETVELD P. - The CBNM Scanning Nuclear Microproble Analytical Facility - Nucl. Instr. and Meth. B69, pp. 463-473, 1992

TECHNICAL EUR REPORTS

TINAGLI F. - ITER Site Licensing Working Party - Licensing of Fusion Plants at JRC Ispra, EUR 14785 EN, August 1992

CONTRIBUTIONS TO CONFERENCES

MALARA C., MENCARELLI T., RICAPITO I., TOCI F., VANSANT E.F. - Characterization of Porous Solids by Analysis of Gas-Physisorption Measurements - presented at the 9th International Zeolite Conference, Montreal, Canada, July 5-10, 1992, (accepted for publication on Microporous Materials)

RICAPITO I., MALARA C., PIERINI G., FACCHINI A., MENCARELLI T., SPELTA B., TOCI F., VIOLA A. - The Adsorption of Gas Mixtures in Zeolites: Experimental Determination and Theoretical Prediction of Equilibria - presented at the 17th SOFT, Roma, 14-18 September 1992

MALARA C., RICAPITO I., SPELTA B., TOCI F., VIOLA A. - Adsorptive Removal of the Impurities from the Exhausted Plasma - Presented to the 9th International Zeolite Conference, Montreal, Canada, July 5-10, 1992, (accepted for publication on Microporous Materials)

MALARA C., PIERINI G., VIOLA A. - The Feasibility of Tritium Extraction Units from Blanket of Fusion Reactors in the Light of Recent Experimental Data - 17th SOFT, Roma, 14-18 September 1992

PERUJO A., ALBERICI S., CAMPOSILVAN J., REITER F. - Hydrogen in the Martensitic Steel Manet: Diffusivity and Solubility Measurements - Presented at the Fourth Topical Meeting on Tritium Technology in Fission, Fusion, and Isotopic Applications - Albuquerque, New Mexico.Fus. Technol. 21 (1992), pp. 800-805, 29 Sep.- 4 Oct., 1991

GIANCARLI L., BARBIER F., FLAMENT T., FUTTERER M., LEROY P., PROUST E., SANNIER J., RAEPSAET X., TERLAIN A., COEN V.,. PERUJO A., SAMPLE T., AGOSTINI P., BENAMATI G. - European Research and Development Programme for Water-Cooled Lithium-Lead Blankets. Present Status and Future Work - Presented at the 10th Topical Meeting on the Technology of Fusion Energy, Boston, Massachusetts, USA, June 7-12, 1992

MANNONE F. - Management of Tritiated Wastes in ETHEL - Proceedings of the 2nd ETHEL/TLK Workshop on Safety Technology in Nuclear Fusion, CEC, JRC-Ispra, 6-7 February 1992, ORA/PRO 36792

REITER F., ALBERICI S., CAMPOSILVAN J., SERRA E., FORCEY K.S., PERUJO A. - Diffusivity and Solubility of Hydrogen Isotopes in the Martensitic Steel DIN 1.4914 (MANET) after thermal exposure at 900 K - presented at the Int. Symposium on Metal Hydrogen Systems, Upsala (Sweden) Jun 8 -12, 1992, to be published in Z. f. Physik Chemie

ROSS D.K., STEFANOPOLUS K.L., FORCEY K.S., IORDANOVA I. - Small Angle Neutron Scattering Studies of H(D) Trapping on Dislocations in Metals - presented at the Int. Symposium Studies on Metal Hydrogen Systems, Upsala (Sweden), June 8-12, 1992 to be published in Z. f. Physik Chemie

FORCEY K.S., PERUJO A., REITER F., LOLLI-CERONI P. - The Formation of Tritium Permeation Barriers by CVD - presented at the 12th Int. Vacuum Congress/8th Int. Congress on Surface Science - The Hague, The Netherlands, 12-16 October, 1992, proceedings will be published in J. Nuclear Materials

PERUJO A., DOUGLAS K., AGOSTINI P., CALDWELL-NICHOLS C.J. - Tritium Permeation through Engineering Components: Jet Bellows Experiments - presented at the 17th Symposium on Fusion Technology, Rome, Italy, 14-18 September, 1992

PERUJO A., CAMPOSILVAN J., REITER F., TOMINETTI S. - ETHEL-001: A Versatile Facility for Plasma-Surface Interaction Research - presented at the IAEA Technical Committee Meeting on: 'Atomic and Molecular Data for Fusion Reactor Technology', Cadarache (France), Oct. 12-16, 1992

TECHNICAL NOTES*

EDWARDS R.A.H., ELEN J.D., BOTTELIER P., FACELLI R. - First project test report: tensile testing of irradiated samples from welds in solution annealed 316L austenitic steel plate - Technical Note No. I. 92.114, Nov. 1992

Industrial hazards

CONTRIBUTIONS TO PERIODICALS AND MONOGRAPHS

ZALDIVAR J.M., HERNANDEZ H., BARCONS C., NOMEN R. - Heat effects due to dilution during aromatic nitrations by mixed acid in batch conditions - (accepted of publication in Journal of Thermal Analysis), 1992

ZALDIVAR J.M., BARCONS C., HERNANDEZ H., MOLGA E., SNEE T.J. - Modelling and optimization of semibatch toluene mononitration with mixed acid from performance and safety viewpoints - Chem. Eng. Sci., 47, pp. 2517-2522, 1992

SNEE T.J., BARCONS C., HERNANDEZ H., ZALDIVAR J.M. - Characterisation of an exothermic reaction using adiabatic and isothermal calorimetry - (accepted for publication in Journal of Thermal Analysis), 1992

ZALDIVAR J.M., PANETSOS F., HERNANDEZ H. - Control of batch reactors using neural networks - Chemical Engineering and Processing, 31, pp. 173-180, 1992

MORRIS S.D., BELL K., OSTER R. - Top-venting of flashing high-viscosity fluids - Chemical Engineering and Processing, vol. 31, pp. 297-305, 1992

BELL K., MORRIS S.D. - Flow visualizations during top-venting from a small vessel, J. Loss Prev. Process Ind., vol 5, no. 3, pp. 160-164, 1992

CONTRIBUTIONS TO CONFERENCES

MORRIS S.D. - Top-venting of flashing high-viscosity fluids: vent-pipe inlet conditions (experiment and theory) - US DIERS Users Group Meeting, Princeton, USA, 30 Sept.-2 Oct. 1992

MORRIS S.D. - Some comments on relief system inlet conditions during two-phase venting - 4th Meeting of ISO Technical Committee 185 (Flashing flow through safety devices), JRC (Ispra), Italy, 7-8 April 1992

MORRIS S.D. - Round-Robin calculation exercise-steam venting problem - 4th Meeting of ISO Technical Committee 185 (Flashing flow through safety devices), JRC (Ispra), Italy, 7-8 April 1992

MORRIS S.D. - Two-phase venting: New formulae for the average vapour drift velocity - US DIERS Users Group Meeting, Princeton, USA, 30 Sept. - 2 Oct. 1992

DUFFIELD J.S., FRIZ G., NIJSING R. - The Venting of Peroxide Solutions - 29th European Two-Phase Flow Group Meeting, Stockholm, June 1992

STAEDTKE H., HOLTBECKER R. - A Hyperbolic Model for Inhomogeneous Two-Phase Flow - 29th Meeting of the European Two-Phase Flow Group, Stockholm, June 1-3, 1992

STATHARAS J.C., BARTZIS J.G., WÜRTZ J. - Prediction of Ammonia Releases Using the ADREA-HF Code - AIChE Summer Meeting, San Diego, (accepted for publication in Process Safety Progress), August 1992

RUEL F., HULD T., SORIA A. - Application of High Resolution Schemes to the Numerical Simulation of Reacting Gas Flows - Quatrième Seminaire sur les ecoulements de fluides compressibles, CEN Saclay, France, Janvier 1992

RUEL F., HULD T. - Simulazione Numerica delle Deflagrazioni/Detonazioni - Seminario sulle Applicazioni della Fluidodinamica Comutazionale, IBM/CISI, 22 gennaio 1992, Milano (I), ORA 36607

HERNANDEZ H., ZLADIVAR J., BARCONS C. - Development of a Mathematical Model and a Numerical Simulator for the Analysis and Optimization of Batch Reacors - ESCAPE 2, CADE & EFCE, 5-7 October 1992, Toulouse (F), Computers & Chemical Engineering, Vol. 17S (1992), pp. 45-51, ORA/ART 37166

HOLTBECKER R., STAEDTE H. - Calcul d'un Ecoulement Diphasique Inhomogène Fondé sur un Système d'Equations Hyperbolique - Quatrième Séminaire sur les Ecoulements de Fluids Compressibles, CEA, 29-31 janvier 1992, Saclay (F), ORA/PRO 36625

BENUZZI A. - Controllo delle Sovrapressioni, Parte 1A: Metodi per il Dimensionamento dei Sistemi di Sfiato - Atti del Corso: Sicurezza dei Reattori Chimici Batch e dei Serbatoi di Stoccaggio, IBM/MATEC, 7-8 maggio 1992, Milano (I), ORA/PRO 36891

BENUZZI A. - Metodi per il Dimensionamento dei Sistemi di Sfiato, Chimica e Sicurezza: Contributo dei Metodi Computerizzati, SChI, 10 giugno 1992, San Donato Milanese (I), ORA 36934

DUFFIELD J. - Emergency Relief: Computational Aspects - Atti della Conferenza sulla Sicurezza dei Reattori Chimici "Batch" e dei Serbatoi di Stocaggio, IBM/MATEC, JRC, 7-8 May 1992, Milano (I), ORA/PRO 36902

TECHNICAL NOTES*

RUEL F. - Numerical Simulation of Reactive Gas Flows: Assessment Calculations of Transonic Multidimensional Flows - CEC/JRC, Technical Note No. I.92.01, January 1992

BENUZZI A. - The JRC Research Project in the Area of Indoor Ventilation and Pollutant Transport Modelling - Technical Note No. I.92.95

HULD T. - Numerical Simulation of Reactive Gas Flows: Assessment Calculations of Deflagration/Detonations, CEC/JRC, Technical Note No. I.92.66, June 1992

RUEL F. - Partial Equilibrium Solvers for Reactive Transonic Flows. Basic Algorithms, CEC/JRC, Technical Note No. I.92.125, November 1992

BEST C., ROEBBELEN D. - TURCOM, Visualization of Reacting Gas Flows on Non-Structured grids - CEC/JRC, Technical Note No. I.92.136, December 1992

Reference methods for the evaluation of structural reliability

CONTRIBUTIONS TO PERIODICALS AND MONOGRAPHS

KAKALIAGOS A., BOUWKAMP J.G. - Experimental study of steel beam-to-column connections under cyclic loads, submitted for publication in the Earthquake Spectra Journal (E.E.R.I), 1992

DONEA J., HUERTA A., CASADEI F. - Finite element models for transient dynamic fluid-structure interaction - in: New Advances in Computational Structural Mechanics, Studies in Applied Mechanics 32, (Ed. P. Ladeveze and O.C. Zienkiewicz), Elsevier, 1992

DONEA J., BELYTSCHKO T. - Advances in Computational Mechanics - Nucl. Eng. Des. 134, pp. 1-22, 1992

DONEA J, QUARTAPELLE L. - An introduction to Finite Element Methods for Transient Advection Problems, Comp. Meths. Appl. Mech. Eng. 95, pp. 169-203, 1992

DONEA J., JONES P., MAGONETTE G., VERZELLETTI G. - The Pseudo-Dynamic Test Method: Recent Advances in Earthquake Engineering and Structural Dynamics- Ouest Editions, Nantes, France, V.E. Davidovici (Ed.), Paris (1992), pp. 769-780, ART 29357

DONEA J., VERZELETTI G., JONES P., PINTO A. (Eds.) - ELSA, European Laboratory for Structural Assessment (Reaction Wall Facility) of the JRC Ispra - Proceedings of the 10th World Conference on Earthquake Engineering, ART 40943

CONTRIBUTIONS TO CONFERENCES

PRAKASH V., POWELL G.H. - DRAIN-2DX a general purpose computer program for dynamic analysis of inelastic structures- - University of California, Berkeley, January 1992

PINTO A.V., NEGRO P., JONES P.M. - Comparative Analysis of R/C Buildings Designed According to EC8 (Frame and Shear Wall) - Proceedings of the 10th World Conference on Earthquake Engineering, Madrid, 1992

NEGRO P., JONES P.M., PINTO A.V. - A Reduced Degree of Freedom Approach in the Pseudodynamic Test Method - Proceedings of the 10th World Conference on Earthquake Engineering, Madrid, 1992

PINTO A.V., NEGRO P., JONES P.M. - Comparative Analysis of R/C Building Structures (Frame Shear Wall) Designed According to EC8 - Proceedings of the 10th WCEE, A.A. Balkema, Madrid, 1992

DONEA J., VERZELETTI G., JONES P.M., PINTO A.V. - The European Laboratory for Structural Assessment (ELSA) - Reaction Wall of the JRC - Ispra, Proceedings of the 10th WCEE, A.A. Balkema, Madrid, 1992

DONEA J., JONES P.M., PINTO A.V. - The European Laboratory for Structural Assessment (ELSA) - Reaction Wall of the JRC - Ispra, Proceedings of the COST-C1 Workshop, Strasbourg, 28-30 October, 1992

NEGRO P., VERZELETTI G. - Indagini Sperimentali nell' Ingegneria Sismica: Il Metodo Pseudodinamico - Atti del Seminario "Evoluzione nella Sperimentazione per le Costruzioni", CIAS, 9-10 ottobre 1992, Ispra (I), ORA/PRO 36998

ALBERTINI C., MONTAGNANI M. - Advanced Impact Testing of Real-Size Thin Sheet Metal Structures for Car Safety - 25th Intern. Symposium on Automotive Technology and Automation, ISATA, 1-5 June 1992, Firenze (I), ORA 36630

BOUSIAS S., VERZELETTI G., FARDIS M., MAGONETTE G. - RC Columns in Cyclic Biaxial Bending and Axial Load - Proceedings of the 10th World Conference on Earghquake Engineering, AEIS, 19-25 July 1992, Madrid (E), ORA/PRO 36766

DONEA J. - Finite Element Methods for Transient Advection Problems - Proceedings of the First National Congress on Computational Mechanics, 3-4 September 1992, Athens (GR) - ORA/PRO 36994

ALBERTINI C., MONTAGNANI M. - Utilization of High Strain Rate Effect for the Strengthening of Advanced Reactor Containment Structures - Proceedings of the ANP Conference, Atomic Energy Soc. of Japan, 25-29 October 1992, Tokyo (Japan), ORA/PRO 37126

ALBERTINI C., MONTAGANI M. - Large Hopkinson's Bar Methods for Advanced Impact Testing of Steel and Concrete Structural Components - Proceedings of the Intern. Symposium on Impact Engineering, 2-4 November 1992, Sendai (Japan), ORA/PRO 37127

PAZIENZA G., PEZZICA G., VIGNOLO M., ALBERTINI C., RODIS A. - Determination of the Armstrong-Zerilli Constitutive Model for AISI 316H Stainless Steel and Application to Cylinder Impact Tests - Proceedings of DYMAT '92, 7-8 October 1992, St. Louis (F), ORA/PRO 37177

NEGRO P. - Effetti delle Vibrazioni del Terreno sugli Edifici - Conferenze su Temi Ambientali, Ass. Pro-Cive, 20 novembre 1992, Caravate (I), ORA 37276

RODIS A., ANDRITSOS F., ALBERTINI C. - Modelling of a "Bicchierino" Type Specimen for Dynamic Biaxial Shear Experiments - Proceedings of the 1st National Congress on Computational Mechanics, GRACM, 3-4 September 1992, Athens (GR), ORA/PRO 36598

GUTIERREZ E., MAGONETTE G., TIRELLI D., VERZELETTI G., FRANCHIONI G. - Real Time and Pseudodynamic Tests on a SDoF Reinforced Concrete Structure - Proceedings of the 10th World Conference on Earthquake Engineering, AEIS, 19-25 July 1992, Madrid (E.), Balkema, Rotterdam (1992), Vol. 6, pp. 3353-3358, ORA/PRO 36773

HUERTA A., CASADEI F., DONEA J. - An arbitrary Lagrangian- Eulerian stress update procedure for coining simulations - 4-th Int. Conf. on Numerical Methods in Industrial Forming Processes, NUMIFORM '92, Sophia Antipolis, France, 14-18 Sept. 1992

TECHNICAL NOTES*

ALBERTINI C., DEL GRANDE A., DELZANO C., KIEFER R., MONTAGNANI M., MURAROTTO M., PIZZINATO E., RODIS A., SCHNABEL W., BLARASIN A. - A New Approach to Crash-Worthiness Studies on Thin Sheet Metal Structures Using the Large Hopkinson's Bar Method - Technical Note No. I.92.133

RODIS A., DEL GRANDE A., MURAROTTO M., PIZZINATO E., SCHNABEL W., ALBERTINI C. - Strain Rate Effects on the Mechanical Properties of Some Thin Sheet Carbon Steels and Glass Fibre Reinforced Composites - Technical Note No. I.92.134

KAKALIAGOS A. - Pseudo-dynamic test-setup of a full scale three-storey one-bay steel frame - Technical Note No. I.92.103, October 1992, C.E.C./ J.R.C, Institute of Safety Technology, 1992

BUCHET P. - Station Graphique Multi-Ecrans - Technical Note No. I.92.144, CEC/JRC, Safety Technology Institute, December 1992

NEGRO P. - Ductility, Damage Indicators and Design - Technical Note, Submitted to the Cooperative Research Group on the Seismic Response of Reinforced Concrete Structures, Ispra, Italy, 1992

CASADEI F. - TPLOT an Interactive Data Management System for Transient Problems, 4th Edition - Technical Note No. I.92.26, March 1992

BOUSIAS S., PINTO A. - Development and Castem 2000 Implementation of Moment-Curvature Relations for Reinforced Concrete Sections Based on a Fibre Formulation - Technical Note No. I.92.07

RODIS A., DEL GRANDE A., PIZZINATO E., SCHNABEL W., ALBERTINI C. - Mechanical Properties of Austenitic Stainless Steel (SUS 304) and Nickel Alloy (Ni 201) under Dynamic Uniaxial and Biaxial Tensile Tests - Technical Note No. I.92.23

DONEA J., JONES P., PINTO A. - European Association of Structural Mechanics Laboratories (ELSA) - 2nd Meeting, 16 October 1992, Technical Note No. I.92.146

SPECIAL PUBLICATIONS

BELLORINI S. - Approccio pseudodinamico all'analisi sismica ed estensione al caso di strutture a massa distribuita - CEC/JRC/AMD, ELSA Laboratory - Ispra (VA), Italy, Oct. 92

PINTO A.V., NEGRO P. - Dynamic Non-linear Analysis of the 4 Storey R/C Building to be Tested in the ELSA-Reaction Wall Facility - Cooperative Research on the Seismic Response of Reinforced Concrete Structures, Interim Report, Lisbon, 1992

JONES P., DONEA J. - Reaction Wall Facility (leaflet) - S.P./I.92.18

Working environment

TECHNICAL NOTES*

NGUYEN H., PETROV-GALERKIN A. - Formulation based on the least-squares finite element concept for multi-dimensional advection-diffusion equation-application to natural convection problems, Technical Note No. I.92.25, JRC-Ispra, March 1992

AL-KHUDHAIRY D.H.A. - Application of the explicit Taylor-Galerkin and implicit Petrov-Galerkin schemes to two-dimensional fluid flow problems, Technical Note No. I.92.46, JRC-Ispra, May 1992

NGUYEN H. - Turbulence models for practical applications: A review, Technical Note No. I.92.67, JRC-Ispra, June 1992

Safeguards

SPECIAL PUBLICATIONS

Meeting on - R&D on Volume and Mass Determination of Liquids in Tanks used at Nuclear Materials Processing Installations - Working document, SPI 92-26, Ispra, Italy, 25-27 March 1992

Parallelisation of large computer codes

TECHNICAL EUR REPORTS*

TAUTGES T.J. - Modelling Fission Product Chemistry and Aerosol Physics on a distributed Memory Parallel Computer: The P-VIC Code - The Commission of the European Communities, EUR 14387, 1992

CONTRIBUTIONS TO CONFERENCES

DELAVAL M., HALLEUX J., MEHR K., TAUTGES J. - Experience with Parallelization of Large Computer Codes - Proceedings of PACTA92, 21-25 September 1992, Barcelona (E), ORA/PRO 36952

MEHR K., TAUTGES T. - Experience with the Parallelisation of Large Computer Codes - PACTA '92, Barcelona, September, 1992

TECHNICAL NOTES*

MEHR K. - Parallelisation of PLEXIS-3C - Technical Note, Dec. 1992

Exploratory research

CONTRIBUTIONS TO PERIODICALS AND MONOGRAPHS

HAGE W., CIFARELLI D.M. - Correlation analysis and neutron count distributions registered updating dead time counter for the assay of fissile materials - Nucl. Sci and Eng. 112, pp.136-158, 1992

ZALDIVAR J.M., PANETSOS F., HERNANDEZ H. - Control of batch reactors using neural networks - Chemical Engineering and Processing, 31, pp. 173-180, 1992

CONTRIBUTIONS TO CONFERENCES

RIEF H., TAKAHASHI H. - A Critical Review of Accelerator-Based Transmutation Systems - invited paper at the OECD-Nuclear Energy Specialists' Meeting on Accelerator-based Transmutation, Wuerenlingen (CH), March 24-26, 1992

TAKAHASHI H., RIEF H. - The Energy Requirement for Transmuting Fission Products - OECD/NEA Second General Meeting of the International Information Exchange Programme on Actinide and Fission Product Separation and Transmutation, ANL, Nov. 11-13, 1992

TAKAHASHI H. RIEF H. - The Energy Requirement for Transmuting and Isolation Fission Products - 1992 ANS/ENS International Meeting, Chicago, Nov. 15-20, 1992

RIEF H., VAN HEUSDEN R., PERLINI G. - Generating Epithermal Neutron Beams for BNCT in Small Reactors - 5th International Symposium on Neutron Capture Therapy for Cancer, Columbus (Ohio), Sept. 12-19, 92

TECHNICAL NOTES*

RICCHENA R. - Research on Optimal Arrangement of the Neutron and Gamma Filter in the HB 11 Channel for Boron Neutron Capture Therapy - Technical Note, October 1992

6.2 MEETINGS

Reactor Safety

ESTER workshop

A Workshop was held at Travedona (VA), Italy, on 22-23 June 1992 on the European Source Term Evaluation Research (ESTER) system code. The Workshop concluded the first development phase of the code architecture performed by two EC contractors under guidance of JRC Ispra. The aims of the meeting were to acquaint potential users and developers with the principal features of the code, to present the first version in a demonstration calculation and to form the ESTER Users Club. There were 35 participants from 13 different Institutes and 6 European countries and as a result of the meeting 10 inscriptions in the Users Club.

FARO European expert group meeting

Experts in the field of LWR severe accident research have been nominated by the EC Member States to assist in the definition of the FARO experimental and analytical programme and in the discussion of the test results. These experts met twice during 1992.

- The second experts group meeting was held at JRC Ispra on 5th February and was attended by 12 representatives of 5 EC Member States.
- The third experts group meeting was held at JRC Ispra on 11th December and was attended by 11 representatives of 5 EC Member States. On request of the European experts, also two US experts (from EPRI and the University of Wisconsin) participated.

LOBI seminar

The LOBI Seminar was held in the Arona (I) Conference Centre from March 31 to April 2, 1992, to provide an open forum for the discussion of major achievements and significant results of the research programme. The Seminar assembled about 60 experts from research organizations of EC member countries, EFTA as well as Central- and East-European countries and international organizations such as the IAEA. It offered also an opportunity for an international exchange of technical and scientific information on current requirements in water reactor safety research. The proceedings of the Seminar were published in the report EUR 14174 EN.

LOBI data users club meeting

The members of the LOBI Data Users Club were convened for their first meeting at JRC-Ispra on October 16, 1992. The meeting was attended by representatives of nine research organizations of EC member countries and offered the opportunity to scientists involved in LOBI post-test analyses to exchange information on the use of the data and related code assessment activities.

Safeguards and fissile material management

Meeting on use of TAME laboratory

A meeting was held at JRC Ispra on 25-27 March 1992 to search advice for the use of the recently set up TAnk MEasurements (TAME) laboratory at Ispra. There were 23 experts, i.e. plant operators and designers, safeguards authorities, etc., from 5 European countries, USA, Japan and Taiwan participating to discuss problems associated with volume/mass measurements in large-scale nuclear plants. The proceedings of the meeting were issued in the working document SPI 92-26.

International workshop on calorimetry

The JRC organized jointly with the EG&G MOUND Applied Technologies and International Workshop on Calorimetry at Ispra on 23-27 March 1992. Developers and users of calorimetric assay instruments and measurements, together with interested persons from national/international inspectorates and nuclear facilities, 23 participants in all, assisted in a one week Workshop on the calorimetric assay of plutonium and tritium. The proceedings of the Workshop were published in EUR 1430 EN.

Fusion Technology and Safety

2ND ETHEL/TLK workshop

This was the 2nd of the annual Workshops held between ETHEL Ispra and the TLK Karlsruhe. The meeting took place in Ispra on 6 and 7th February 1992 with 10 representatives from KfK and 16 from Ispra. The Workshop on the safety technology in thermonuclear fusion concentrated on tritium supply, accountancy, diagnostics and instrumentation as well as on waste management criteria, safety aspects, commissioning procedures and future prospective. The proceeding will be published as a JRC-SP report shortly.

Tritium technology and safety meetings

A series of joint and individual meetings were held at Ispra on 21-23 September 1992 with representatives from Canada, USA and Japan on aspects of tritium technology and safety. One of the objectives of the meeting was to discuss the mutual interest for a collaboration in ETHEL. Detailed terms of collaboration were discussed with USDOE and TSTA.

Industrial Hazards

Course on chemical reactors safety

A course has been organized in Ispra and in collaboration with IBM and MATEC Milan (I) on 7-8th May 1992 on the safety of chemical batch reactors and of storage tanks. 28 Representatives from the Italian chemical industry participated.

DECHEMA working group meeting

A meeting of the DECHEMA Working Group took place in Ispra on 21-22 September 1992 on the reactor techniques of difficult processes in view of safety technology. It was the 15th meeting of the DECHEMA-GWC which meets about 1 to 2 times per year, normally in Germany, to discuss industrial problems. This meeting was organised at Ipsra to establish contacts to the group. On the second day the STI programme on industrial hazards was presented. About 15 representatives of the industry mainly from Germany participated.

Reference Methods for the Evaluation of Structural Reliability

Workshop on pre-normative research for EUROCODE 8

A Workshop was jointly organized by the "Laboratorio Nacional de Engenharia Civil" (P) and JRC Ispra at Lisbon on 22-23rd June 1992 on the pre-normative research for the Eurocode 8. The Workshop was attended by about 30 participants representing DG III of the European Commission, the Technical Committee CEN-TC 250 in charge of drafting Eurocode 8 and the European Association of Structural Mechanics Laboratories representing 10 European countries and the Safety Technology Institute.

Seminar on evolution in experimentation for constructions

A Seminar, taking place at Ispra on 9-10th October 1992, was jointly organized by the "Centro Internazionale di Aggiornamento Sperimentale-Scientifico" (I) and JRC Ispra on the evolution in experimentation for constructions. This two-day Seminar was attended by about 50 participants, mainly from Italy, Greece and the JRC. The central subject for the meeting was a discussion of recent progress in experimental methods in civil engineering. The Seminar proceedings will be published by CIAS.

Inauguration of ELSA

The reaction wall facility ELSA was inaugurated in an official opening ceremony on the Ispra site on 16th October 1992 and in the presence of the Vice-President F.M. Pandolfi of the European Commission. This event was attended by more than 300 persons representing Authorities and the research Community from the Community Member States. Demonstration tests were performed after the academic part of the ceremony to illustrate the testing potential of ELSA. In the afternoon, a meeting of the European Association of Structural Mechanics Laboratories took place with the aim of reviewing progress in collaborative research sponsored by the JRC and discussing future opportunities, such as the setting up of a collaboration network in the framework of the programme on Human Capital and Mobility.

7 LIST OF TABLES AND FIGURES

7.1 LIST OF TABLES

Reactor safety

Management and storage of radioactive wastes

Safeguards and fissile material management

Fusion technology and safety

Safeguards

Parallelisation of large computer codes - support to DG XIII

Exploratory research

Structure and distribution

The ELSA reaction-wall facility

LIST OF FIGURES

Reactor Safety

Working environment

Safeguards

Exploitation of new instrumentation value programme - developments

NDA detection limits for plutonium in radioactive waste

Iron ore reduction

Purification of the gaseous feed stream to fuel cell

Neural networks for chemical reactor control

Study of the true stress-strain response of plain concrete by a bundle Hopkinson's bar technique

Structure and distribution

The ELSA reaction-wall facility

PETRA

PERLA

ETHEL

8 INDEX OF AUTHORS

Editors

Fasoli-Stella P.
Krischer W.

Reactor Safety

Addabbo C.
Benuzzi A.
Bonanni E.
Capitaõ J.A.
Delaval M.
De Santi G.F.
Drossinos Y.
Fasoli-Stella P.
Fermandjian J.
Hohmann H.
Jones A.V.
Magallon D.
Markovina A.G.
Rief H.
Shepherd I.M.
Von der Hardt P.
Wider H.U.
Worth B.
Yerkess A.

Safeguards and Waste

Barbera G.
Bertelli S.
Dierckx R.
Dworschak H.
Franklin M.
Guardini S.
Hage W.
Hunt B.A.
Krüger G.
Pedersen B.
Schillebeeckx P.
Tinagli F.
Vocino V.

Fusion Technology and Safety

Dworschak H.
Edwards R.A.H.
Forcey K.
Malara C.
Mannone F.
Modica G.
Perujo A.
Pierini G.
Reiter F.
Ricapito I.
Toci F.
Vassallo G.
Viola A.

Industrial Hazards

Andronopoulos S.
Bassani C.
Busse C.A.O.
Duffield J.S.
Krüger G.
Morris S.D.
Nijsing R.
Ruel F.H.M.
Städtke H.
Würtz J.
Zaldivar J.M.

Reference Methods for the Evaluation of Structural Reliability

Anthoine A.
Buchet P.
Casadei F.
Donea J.M.
Gutierrez E.
Halleux J.P.
Kakaliagos A.
Maglia E.
Magonette G.
Negro P.
Papa L.
Pegon P.
Pinto A.
Renda V.
Roveda L.
Saldarini G.
Soria A.
Schnabel W.

Working Environment

Support to Community Policies

Association of Laboratories

Exploratory Research

Large Test Facilities

Acknowledgements

The editors would like to thank Ms. E. Della Chiesa, Mrs. K. Vasconi and Mrs. H. van Paemel for text editing, the Public Relations and Publications Office for its assistance and, in particular, Mr. J.N. Wells for the graphical layout.

9 GLOSSARY OF ACRONYMS AND ABBREVIATIONS

ACE	Advanced Containment Experiments
ADECO	Atelier de Demantelement Element Combustible ORGEL
AEA	Atomic Energy Agency
AECC	Active Euratom Coincidence Counter
AECL	Atomic Energy of Canada Laboratory
AEE	Atomic Energy Establishment
AFW	Auxiliary Feed Water
AISI	austenitic steel
ALE	Arbitrary Lagrangian - Eulerian
AMIW	All Melt In Water
AN	ANtech Technology Ascot
ANL	Argonne National Laboratory
ASCII	American Standard Code for Information Interchange
AWCC	Active Well Coincidence Counter
BCT	Base Case Test
BETHSY	Boucle d'Etudes Thermohydrauliques SYsteme
BNCT	Boron Neutron Capture Therapy
BNFL	British Nuclear Fuel Ltd
BP	British Petroleum
Bw	Boiling water
BWR	Boiling Water Reactor
CABRI	test reactor at Cadarache
CAL	CALorimetry
CALDEX	CALibration Demonstration EXercise
CARN	Chemical Abstract Registry Number
CBNM	Central Bureau for Nuclear Measurements
CCFL	Counter Current Flow Limitation
CCM	Corium Coolant Mixing
CEA	Commissariat a l'Energie Atomique
CEC	Commission of the European Communities
CEN	Centre d'Etudes Nucleaires
CENG	Centre d'Etudes Nucleaires de Grenoble
CF	Coupling Factor
CFC	Carbon Fiber Composite
CIAS	Centro Italiano di Aggiornamento Scientifico
CIEMAT	Centro de Investigaciones Energeticas Medio-Ambientales y Tecnologicas, Madrid
CISI	Compagnie Internationale Services Informatiques
CORA	Complex Out-of-pile Rod bundle Assembly
COST-C1	European Concerted Action on control of semi-rigid behaviour
CPT	Counter - Part Test
CPU	ComPUter
CRIEPI	Central Research Institute of Electric Power Industry
CRIS	Centro di Ricerca Idraulica e Strutturale
C/S	Containment/Surveillance
CSNI	Committee on the Safety of Nuclear Installations
CSTF	Containment System Test Facility
CUC	CATHARE Users'Club
CUS	Clean - Up System
DA	Destructive Analysis
DCH	Direct Containment Heating
DCS	Direction Contrôle de Sécurité
DEC	Digital Equipment Company
DECHEMA	DEutsche Gesellschaft für CHEMisches Apparatewesen, Chemische Technik und Biotechnologie e.V., Frankfurt
DEMO	DEMOnstration power reactor
DG	Directorate General
DIERS	Design Institute for Emergency Relief Systems
DIN	Deutsche Industrie - Norm
DISP	DIrezione Sicurezza Nucleare e Protezione Sanitaria
DoF	Degree of Freedom
DOS	Disk Operating System
DRACULA	Depressurization, Relief And Containment Using Large Apparatus
DGT	Differential Thermal Gravimetry
EAC	European Accident Code
EASML	European Association of Structural Mechanics Laboratories
EB	Electron Beam
EC	European Communities
EC 8	EuroCode N° 8
ECCS	Emergency Core Cooling System
EdF	Electricité de France
EDF	Experimental Data File
EEDF	ELSA Experimental Data Files
EFTA	European Free Trade Association
ELSA	European Laboratory for Structural Assessment
EM	Enrichment Meter
EMI	Enrichment Meter Inspection
ENEA	Energia Nucleare ed Energie Alternative
ENEL	Ente Nazionale per Energia ELettrica
ENUSA	Empresa Nacional de Uranio S.A.
EOB	End of Break-up
EPRI	Electric Power Research Institute
ERCOFTAC	European Research Community On Flow, Turbulence And Combustion
ESD	Euratom Safeguards Directorate
ESTER	European Source Term Evaluation Research
ETHEL	European Tritium Handling Experimental Laboratory
EUREF	EUropean REFerence code
FAL	FALcon

FARO	Fuel melting And Release Oven
FBFC	Franco-Belge Fabrication Combustible
FCI	Fuel Coolant Interaction
FE	Finite Element
FIFO	First-In-First-Out
FIRES	Facility for Investigating Runaway Events Safely
FISIM	FIRES SIMulator
FLADIS	two-phase FLAshing releases DISpersion
FP	Fission Product
FPT	Fission Product Test
FVS	Flux Vector Splitting
FWP	FrameWork Programme
FZR	Forschungs-Zentrum Rossendorf
GC	Gas Chromatography
GNS	Gesellschaft für Nuclear Service
GRS	Gesellschaft für Reaktor-Sicherheit
HCM	Human Capital and Mobility
HEU	Highly Enriched Uranium
HFR	High Flux Reactor
HLNCC	High Level Neutron Coincidence Counter
HP	High Purity
HPGe	High Purity Germanium detector
HPIS	High Pressure Injection System
HRGS	High Resolution Gamma Spectrometry
IAEA	International Atomic Energy Agency
IAM	Institute of Advanced Materials
IBM	International Business Machines
IGNITOR	IGNItion TOrus
IKE	Institut für Kerntechnik und Energiesysteme
I/O	Input/Output
IPSN	Institut de Protection et de Sûreté Nucléaire
IR	InfraRed
ISEI	Institute for Systems Engineering and Informatics
ISO	International Standards Organisation
ISP	International Standard Problem
ITER	International Thermonuclear Experimental Reactor
JAERI	Japan Atomic Energy Research Institute
JET	Joint European Torus
JRC	Joint Research Centre
KfK	Kernfoschungszentrum Karlsruhe
KFKI	Hungarian Central Research Institute for Physics
KROTOS	small-scale steam explosion facility, Ispra
KWU	KernkraftWerks-Union
LACE	LWR Aerosol Containment Experiments
LAN	Local Area Network
LANL	Los Alamos National Laboratory
LDTF	Large Dynamic Test Facility
LER	Laboratory for Exploratory Research
LEU	Low Enriched Uranium
LIBRETTO	LIquid BReeder Experiment with Tritium Transport Option
LLNL	Lawrence Livermoore National Laboratory
LMFBR	Liquid Metal Fast Breeder Reactor
LNEC	Laboratorio Nacional de Engenharia Civil
LO-AX	ORTEC code for an n-type HP Ge-detector
LOBI	LWR Off-normal Behaviour Investigations
LOCA	Loss Of Coolant Accident
LOFT	Loss Of Flow Test
LOFW	Loss of Feed Water
LPIS	Low Pressure Injection System
LSTF	Large - Scale Test Facility
LWR	Light Water Reactor
MANET	MArtensitic for NET
MBC	Melt - Bottom Contact
MCCI	Molten Core - Concrete Interaction
MCNP	MonteCarlo Neutron Photon transport code
MG	Multi - Group
MGA	MultiGroup Analysis
MLP	Mordenite Large Port
MOX	Mixed OXyd
MPMC	MultiPhase - MultiComponent
MTR	Materials Testing Reactor
MUF	Material Unaccounted For
MWC	Melt - Water Contact
NCS	Network Computing System
NDA	Non - Destructive Analysis
NEA	Nuclear Energy Agency
NET	Next European Torus
NFC	Nuclear Fuel Cycle
NMR	Nuclear Magnetic Resonance
NN	Neural Network
NPT	Non - Profileration Treaty
NUPEC	NUclear Power Engineering Center, Japan
OECD	Organisation for Economic Cooperation and Development
OLAM	On - Line Aerosol Monitor
OMEGA	Options of Making Extra Gains from Actinides
PBF	Power Burst Facility
PC	Personal Computer
PECC	Passive Euratom Coincidence Counter
PERLA	PERformance LAboratory for safeguards
PETRA	Plant for Evaluation and Testing of Radwaste management Alternatives
PHONID	PHOto Neutron Interrogation Device

PIA	Pulse Interval Acquisition
PILC	Pillared Inter-Layer Clay
PIMS	Plutonium Input Measuring Station
PIV	Physical Inventory Verification
PNC	Passive Neutron Counting
PQ	Philadelfia Quarz
Pr	Prandtl number
PSA	Pressure Swing Adsorption
PSD	PSeudo - Dynamic
PUPA	PlUtonium Pin Assay
PWR	Pressurized Water Reactor
QA	Quality Assessment
QC	Quality Control
QT	Quenching Test
Ra	Raleigh number
RAM	Random Access Memory
RAPID	Ricerca Automatica di Parole In Data bank
R/C	Reinforced Concrete
RCWG	R/C Working Group
R & D	Research and Development
RDF	Reduced Degree of Freedom
RTF	Radioiodine Test Facility
RWTH	Rheinland - Westfälische Technische Hochschule, Aachen
SAT	Spectrum Analysis Tools
SAWG	Scientific Analysis Working Group
SCA	Shared Cost Action
SD	Standard Deviation
SEISMIER	Studies and Experimental Investigations on Structural Model to Improve Earthquake Resistance
SFD	Severe Fuel Damage
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SIET	Società Informazioni Esperienze Termoidrauliche
SMA	Shielded Metal Arc
S - N	discrete ordinate method
SNL	Sandia National Laboratories
SPECTRA	Safeguards PErla Centre for TRAining
SPEFC	Solid Polymeric Electrolyte Fuel Cell
SPES	Simulatore ESPErienze Sicurezza
SS	Stainless Steel
ST	Scoping Test
STEP	Science and Technology for Environmental Protection
STI	Safety Technology Institute
STORM	Simplified Tests On Resuspension Mechanism
SUN	Micro-Systems Computer Corporation
TAME	TAnk MEasurement laboratory
T & C	Testing and Commissioning
TCA	Time Correlation Analyser
TG	Technical Group
TIMO	TIMOshenko element with linear approximation
TK	Temperatur Korrelation
TLK	Tritium Laboratory Karlsruhe
TRIGA	TRaInIng reactor General Atomic
TSTA	Tritium System Test Assembly
TUI	TransUran Institut
TZM	molybdenum alloy
UHV	Ultra - High Vacuum
UK	United Kingdom
UOP	Union Carbide
UPC	Universitat Politecnica de Catalunya
UPM	Universidad Politecnica de Madrid
US	United States
USDOE	United States Department Of Energy
USNRC	United States Nuclear Regulatory Commission
UTS	UlTrasonic Sensor
UV	UltraViolet
VTT	Valtion Teknillinen Tutkimuskeskus, Finland
WCEE	World Conference on Earthquake Engineering
WDB	Working on Data Base
Z	atomic number

APPENDIX

A LARGE TEST FACILITIES

THE ELSA REACTION-WALL FACILITY

In 1992, the Safety Technology Institute has completed the construction of a structural assessment laboratory based on a 16m high, 21m wide reaction wall (*Fig. A1.1*). This new facility, now named ELSA (European Laboratory for Structural Assessment), will make it possible to study how large-scale models of complex civil engineering structures react when subjected to severe static or dynamic loading. Designed to resist the huge forces, typically several hundred tonnes, which are necessary to deform and seriously damage full-scale test models of structures, the ELSA reaction-wall is actually one of the largest facilities of its type in the world, only exceeded by Japan.

In addition to static and cyclic tests on large structures and components, the facility is equiped to perform the so-called pseudo-dynamic (PSD) test technique, enabling for example, the simulation of earthquake loading on full-scale buildings.

Fig. A1.1 ELSA Reaction-Wall and three-storey test frame

Basis of the pseudo-dynamic test method

A pseudo-dynamic test is a test which, although carried out quasi-statically, uses on-line computer calculation and control together with experimental measurement of the actual properties of the structure to provide a realistic simulation of the dynamic response. The equations of motion for a discrete parameter model of the test structure are solved on-line using a step-by-step numerical integration method. Inertial and viscous damping forces are modelled analytically - a straightforward matter, whilst the nonlinear structural restoring forces, which are virtually impossible to model accurately, are measured experimentally. This process automatically accounts for the hysteretic damping due to plastic strain and damage of the structural materials which is the major source of energy dissipation.

Fig. A1.2 illustrates the method schematically as applied to an earthquake simulation test of a civil engineering structure. A record of an actual or artificially generated earthquake ground acceleration history is given as input data to the computer and the horizontal displacements of the floors (the levels at which the mass of the building can be considered to be concentrated) are calculated for a small time-step. These displacements are then applied to the structure by servo-controlled hydraulic actuators attached to the reaction wall. Load-cells on the actuators measure the forces necessary to achieve the required deformation (the structural restoring forces) and these are then used in the next step of the calculation.

Because the dynamic inertia forces are modelled, there is no need to perform the test in the real timescale and typically an earthquake lasting of the order of ten seconds in real time, is simulated in a pseudo-dynamic test in about an hour. Herein lies one of the major advantages of the method. It is possible to test very large models with only a modest hydraulic power requirement. This is in contrast with the more conventional "shaking-table" technique which, although running in real time, is restricted to the testing of components or small-scale models of large structures. The second major advantage compared to a shaking-table is the possibility to monitor very closely the progression of damage in the structure and to stop at any moment for a detailed examination. The two test methods are in fact complementary. Shaking-tables are used for preliminary tests and parameter studies at small scale, or when rate-dependent materials or

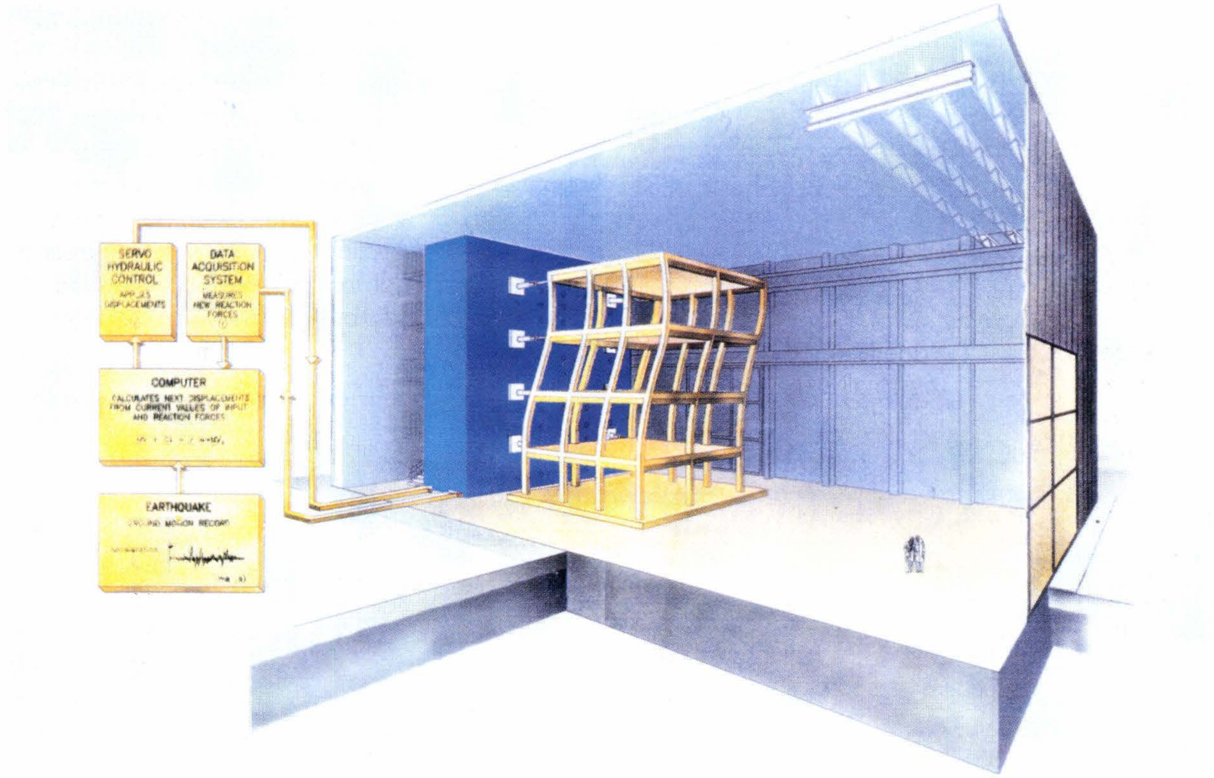

Fig. A1.2 Pseudo-dynamic test (schematic)

distributed-mass structures are involved. Pseudo-dynamic tests are especially useful for confirmatory tests at full scale where the exact material and construction details can be reproduced, or when multi-point loading is required.

The potential of the PSD test method has not yet been fully exploited and new fields of application can be expected. The JRC installation is the first to use fully digital servo control of the applied displacements, allowing a highly accurate test procedure and a much more versatile use of the various mathematical algorithms for numerical time integration of the equations of motion.

By using a mathematical technique, known as sub-structuring significant further developments are possible. With such a procedure only the part of a structure of particular interest is tested experimentally, whilst the rest is modelled analytically. The pilot computer can then account for the interactions between the two parts of the structure in calculating the displacements to be imposed on the tested part. Thus, structures much larger than the laboratory itself, such as for example a bridge, can be tested. Here, by assuming elastic behaviour of the bridge deck, the pilot computer can account for the effects of its behaviour in calculating the displacements to be imposed on the piers and the physical testing can be limited to the piers alone.

Alternatively, when rate-dependency of structural materials is important, much faster testing speeds can be achieved by reducing the size of the physical model to just those few components expected to show non-linear behaviour whilst the rest, which behave linearly, is simulated in the computer. Soil-structure effects can also be taken into account by this means providing a suitable analytical model of the soil behaviour is available.

Technical data for the reaction-wall system

The details of the reaction-wall/strong-floor system are given in *Fig. A1.3* and *Table A1.1* below. The dimensions allow the testing of full-scale buildings up

A.3 PERLA

The PERformance LAboratory (PERLA) is a complex of different laboratories located at the JRC in the Nuclear Island of STI. The decision to set-up PERLA was taken by the JRC Directorate in conformity with the obligations of subsidiarity required by the Commission of European Communities.

The basic aim of PERLA is to improve technology transfer from laboratory development to the application of safeguards instruments and techniques in an industrial environment.

Safeguard laboratory and plant operators either from inside or outside the Communities as well as from national and international inspectorates can find in PERLA a suitable location and adequate tools to calibrate their instruments, to follow training courses and to assess the performances of their techniques.

The main tasks of PERLA are currently:

- Performance Evaluation of Non-Destructive Analysis (NDA) techniques,
- Preparation and characterization of standards for NDA,
- Development of customer oriented systems and software,
- Training and technology transfer,
- Basic research.

From 1987 to 1992 the above tasks were carried out in the PRE-PERLA laboratory which was realised prior to the construction of New-PERLA (*Fig. A 3.1*).

The New-PERLA licensing tests started beginning 1992. Combined tests of ventilation, physical protection and health physics systems have been performed by the Nuclear Experiment Division (NED) licensing and operative groups, under the supervision of the Italian Licensing Authority (ENEA-DISP). The nuclear tests have been carried out in July 1992. The facility has started operation in August 1992. A more detailed description of PERLA has already been given in the annual report 1991 [1], Appendix A2.

References

[1] Safety Technology Institue, Annual Report 1991, Joint Research Centre Ispra, EUR 14803 EN

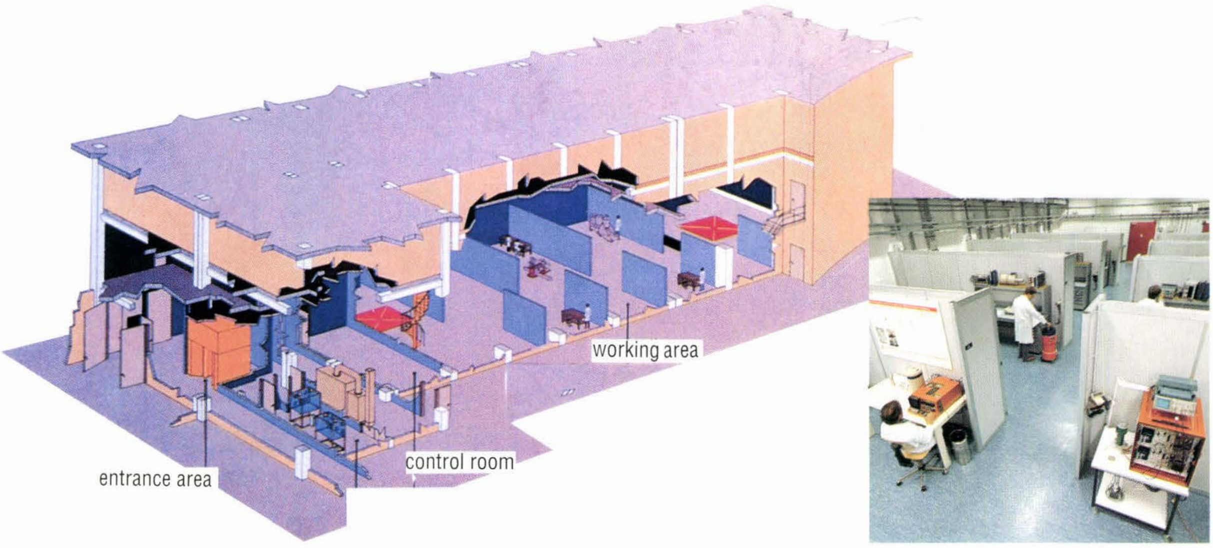

Fig. A3.1 View of the new-PERLA facility showing the internal layout

A.4 ETHEL

The European Tritium Handling Experimental Laboratory (ETHEL) is a new tritium research facility located at the JRC-Ispra in the Nuclear Island of STI. The laboratory, envisaged to be licenced for 37 PBq (1 MCi or 100 g) of tritium and commissioned with tritium in autumn 1993, embraces a volume of approximately 1300m3 and can host a total of about 25 people.

The operation of the laboratory itself and the future experimental activities planned represent a direct contribution of the European Communities to the European Fusion Programme and ultimately to the worldwide efforts in the frame of the ITER project.

A detailed description of systems for managing, containing and processing tritium in ETHEL as well as for conditioning and storing ETHEL tritiated wastes has already been presented in Appendix A3 of ref. [1]. *Fig. A 4.1* gives a photographic view of the ETHEL building.

References

[1] Safety Technology Institute, Annual Report 1991, Joint Research Centre Ispra, EUR 14803 EN

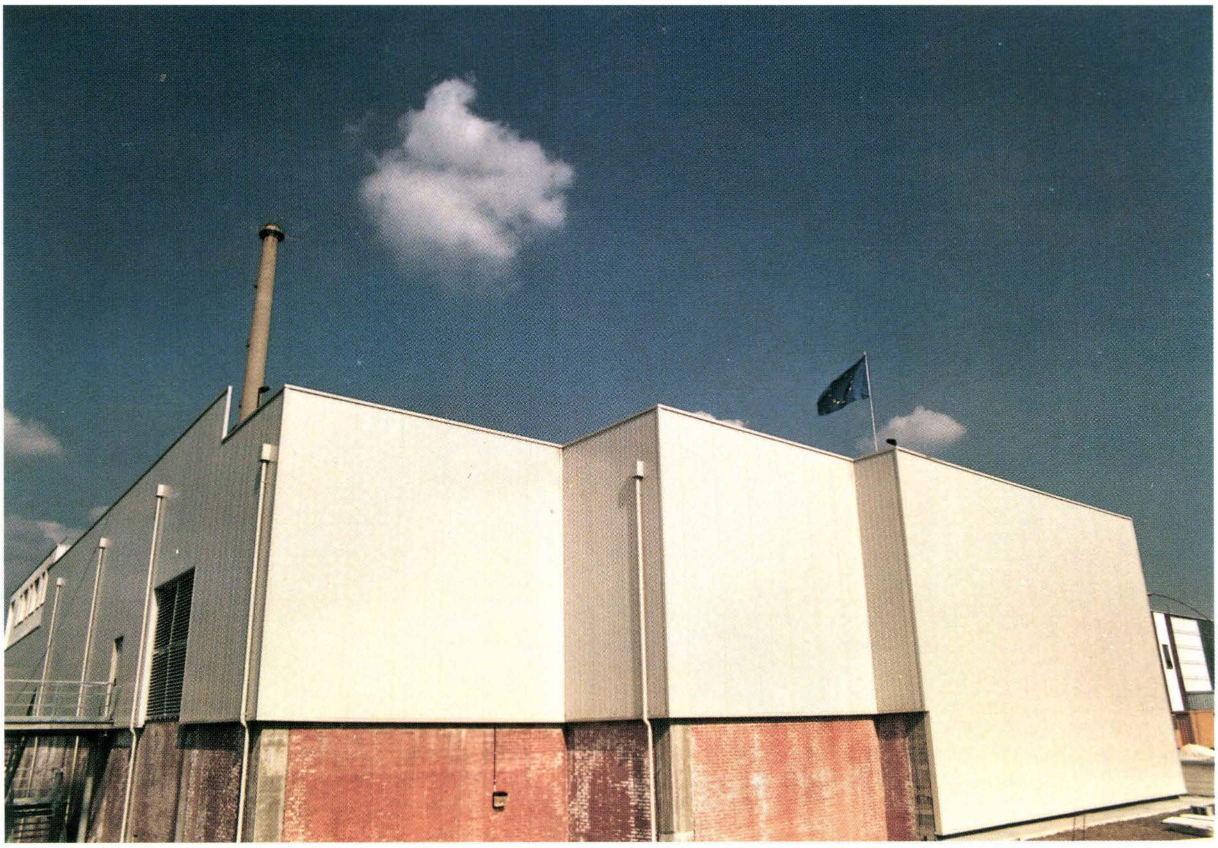

Fig. A4.1 View of the ETHEL building